50-331

DRAFT SUMMARY
OF
EQUIPMENT QUALIFICATION PROGRAM STATUS
FOR THE
DUANE ARNOLD ENERGY CENTER

ATTACHMENT 1

Technical Evaluation Report

RESPONSE SUMMARY

RESOLUTION OF QUALIFICATION DEFICIENCIES IDENTIFIED IN NRC TECHNICAL EVALUATION REPORT 42488

MOTORIZED VALVE ACTUATORS

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
01, 02, 05, 06 08, 14, 16, 121, 122, 124	Limitorque Motorized Valve Actuators (Qty 46)	IIa	1) Documentation Insufficient	Required documentation on file showing qualification.
			2) Similarity of Equipment and test specimen not established	Summary analysis establishing similarity in EQR Section VII A.
			3) Qualified life not established	Aging analysis summary in EQR Section VII A establishes qualified life.
03, 04, 13, 17 123	Limitorque Motorized Valve Actuators (Qty. 29)	IIc	1) Qualified life not established	Aging analysis summary in EQR Section IX A establishes qualified life.
			2) Documentation Insufficient	Required documentation on file showing qualification
07, 09, 10, 11 18	Limitorque Motorized Valve Actuators (Qty. 14)	Ib	1) Documentation Insufficient	Documentation of motor brake subassembly not available. Replacement with qualified components scheduled during the next outage. See EQR Section II Action Item 19 for description and justification for continued operation.
			2) Similarity of Equipment and test specimen not established	
			3) Qualified life not established	
12, 15	Limitorque Motorized Valve Actuators (Qty. 4)	Ib	1) Documentation Insufficient	Required documentation on file showing qualification.
			2) Qualified life not established	Aging analysis summary in EQR Section VI B establishes qualified life.

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MOTORIZED VALVE ACTUATORS

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
19	Electrodyne Motorized Actuator (Qty 2)	1b	1) Documentation Insufficient	Documentation of motor and Valve motor brake not available. Replacement with qualified components scheduled during the next refuel outage. See EQR Section II Action Item 23 for description and justification for continued operation.

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SOLENOID VALVE ACTUATORS

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
20, 21, 23, 28 36, 40, 41, 42 43, 44, 45, 46	Asco Solenoid Actuated Valve (Qty. 45)	Ia	None	Equipment Qualified
25, 29, 31, 32 33, 34	Asco Solenoid Actuated Valve (Qty. 19)	Ib	1) Documentation Insufficient	Equipment replaced by fully qualified components during refuel outage of 1983. See EQR Section III Action Item 24 for description.
26, 27, 30, 35 37, 38, 39	Asco Solenoid Actuated Valve (Qty. 19)	IIc	1) Aging Analysis inadequate 2) Qualified life not established	Aging analysis summary establishing qualification and qualified life in EQR Section IX B
24	AVCO Solenoid Actuated Valve (Qty. 4)	IV	1) Aging analysis inadequate 2) Qualified life not established 3) Aging degradation program inadequate 4) Spray criteria inadequate	Aging analysis documentation on file. Summary establishing qualification and qualified life in EQR Section VIII A. Spray analysis showing qualification in EQR Section VIII A
47, 48	Asco Solenoid Actuated Valve (Qty. 16)	IV	1) Documentation referenced not made available	Documentation on file establishing establishing qualification. summary of documents in EQR Section VIII B

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MOTORS

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
88	Westinghouse Blower Motor (Qty. 2)	Ib	1) Documentation insufficient	Adequate documentation of radiation qualification not available. Replacement with qualified component scheduled during the next refuel outage. See EQR Section II Action Item 14 for description and Justification for Continued Operation.
89	Seimens Blower Motor (Qty. 2)	Ib	1) Documentation insufficient	Documentation on file establishing qualification. Radiation and analysis completed and test data confirms qualification. See EQR Section II Action Item 12.
90	Louis Allis Fan Motor (Qty. 2)	IV	1) Documentation referenced not made available	Required documentation on file establishing qualification. Summary in EQR Section VIIIE.
91, 119	General Electric Pump Motor	IV	1) Documentation referenced not made available	Required documentation on file establishing qualification. Summary in EQR Section VIIIF.

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DETECTORS, SENSORS, PRIMARY ELEMENTS

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
22, 68	Victoreen Radiation Element (Qty. 4)	Ia	None	Equipment Qualified.
67	General Electric Radiation Element (Qty. 2)	IIa	1) Documentation Insufficient	An analysis of functions has led to a reclassification of Exempt from qualification. Component not required for primary accident monitoring. See EQR Section VII 0 for evaluation.
57	DeLaval Level Element (Qty. 2)	IIa	1) Documentation Insufficient	An analysis of functions has led to a reclassification of Exempt from qualification. Component not required for primary accident monitoring. See EQR Section VII 0 for evaluation.
95, 96, 97	Leeds & Northrup Temperature Elements (Qty. 16)	IIa	1) Documentation Insufficient	Documentation establishing qualification not available. Replacement with qualified components scheduled during the next refuel outage. See EQR Action Item 33 for description and Justification for Continued Operation.
102	Rosemount Temperature	Ib	1) Documentation Insufficient	Documentation establishing qualification not available. Replacement with qualified components scheduled during the next refueling outage. See EQR Action Item 25 for description and Justification for Continued Operation.

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DETECTORS, SENSORS, PRIMARY ELEMENTS

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
92, 93, 99 100, 101	NECI, General Electric, Leeds & Northrup Temperature Elements (Qty. 17)	IIa	1) Documentation Insufficient	An analysis of functions has led to a reclassification of Exempt from qualification. Component not required for primary accident monitoring. See EQR Section VII O for evaluation.
98	Gulton Temperature Elements (Qty. 4)	IV	1) Documentation referenced	Required documentation on file. Technical summary of documents in EQR Section VIII G showing qualification.
104, 105	NECI Temperature Elements (Qty. 15)	IIa	1) Documentation Insufficient	Documentation evaluation establishing qualification summarized in EQR Section VII H.
106	Pyco Temperature Element	IIa	1) Similiarity of equipment and test specimen not established	Analysis establishing similiarity in EQR Section VII I.
			2) Aging degradation Program inadequate	Aging analysis summary establishing qualification and qualified life in EQR Section VII I.
			3) Qualified life not established	
			4) Test failures or anomalies not resolved	Evaluation of test results establishing simliiarity of successful test specimen and equipment documented in EQR Section VII I.

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Attachment 1
NG-84-1051

RESOLUTION OF QUALIFICATION DEFICIENCIES IDENTIFIED IN NRC TECHNICAL EVALUATION REPORT 42488

SENSOR SWITCHES

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
49	Barton 289 Pressure Actuated Switch (Qty. 1)	IIa	1) Documentation Insufficient	Analysis establishing similiarity of equipment and qualified test specimen summarized in EQR Section VII B. Aging analysis establishing qualification and qualified life in EQR Section VII B.
60	Pressure Controls Pressure Actuated (Qty. 24)	IIa	1) Aging degradation program inadequate 2) Qualified life not established	Aging analysis establishing qualification and qualified life in EQR Section VII L.
			3) Spray criteria inadequate	Site inspection has verified that this equipment is not subject to direct spray. See EQR Section II Action Item 38 for description.
61, 62	Static-O Ring Pressure Actuated Switch (Qty. 12)	IV	1) Documentation referenced not made available	Required documentation on file establishing qualification. Document summary in EQR Section VIII C.
50	Barton 289 Presure Actuated Switch (Qty. 4)	Ib	1) Documentation Insufficient	Analysis establishing similiarity of equipment and qualified test specimen summarized in EQR Section VII C. Aging analysis establishing qualification and qualified life in EQR Section VI C. See EQR Section II Action Item 17, 18 for description of resolution.

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NG-84-1051

RESOLUTION OF QUALIFICATION DEFICIENCIES IDENTIFIED IN NRC TECHNICAL EVALUATION REPORT 42488

SENSOR SWITCHES

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
63	Barksdale Pressure Actuated Switch (Qty. 4)	Ib	1) Documentation Insufficient	An analysis of function has led to reclassification of exempt from qualification. Component not required for primary accident monitoring. Operational function of MSIV LCS is monitored by qualified flow sensors. See EQR Action Item 34 for description.
65	Barton Pressure Actuated Switch (Qty. 2)	IV	1) Documentation referenced not made available	Required documentation on file establishing qualification. Document summary in EQR Section VIII D.
56	Delaval Level Activated Switch (Qty. 2)	IIa	1) Documentation Insufficient	An analysis of function has led to reclassification of exempt from qualification. Component not required for primary accident monitoring. See EQR Section VII O for evaluation.
66	Fenwal Temperature Controller (Qty. 8)	IIa	1) Documentation Insufficient	Components are in an area harsh for radiation only. Radiation susceptible parts were relocated to an area acceptable for radiation.. An analysis of aging establishes qualification. See EQR Section VII D for description.
94	Essex Controls Temperature Switch (Qty. 4)	Ib	1) Documentation Insufficient	An analysis of thermal and radiation aging establishes qualification. Documentation on file. See EQR Section II Action Item 27 for description.

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SENSOR SWITCHES

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
103	Penn Temperature Switch (Qty. 2)	Ib	1) Documentation Insufficient	Documentation establishing qualification not available. Replacement with qualified components scheduled during the next refueling outage. See EQR Section II Action Item 25 for description and Justification for Continued Operation.

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INSTRUMENT TRANSMITTERS

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
51	SK Instruments Flowmeter (Qty. 4)	Ib	1) Documentation Insufficient	Documentation establishing qualification not available. Replacement with qualified components scheduled during the next refueling outage. See EQR Section II Action Item 10 for description and Justification for Continued Operation.
52	General Electric Flow Transmitter (Qty. 2)	IIa	1) Documentation Insufficient	An analysis of function has led to a reclassification of exempt from qualification. Component not required for primary accident monitoring. See EQR Section VII O for evaluation.
53, 54, 120	GE, Barton Pressure Transmitter (Qty. 5)	IIa	1) Documentation Insufficient	An analysis of function has led to a reclassification of exempt from qualification. Component not required for primary accident monitoring. See EQR Section VII O for evaluation.
55	Barton 763 Level Transmitter (Qty. 2)	IIa	1) Documentation Insufficient	Required documentation on file establishing qualification. Aging analysis summary in EQR Section VII C establishes qualified life.
58, 59	Barton 764 Pressure Transmitter (Qty. 4)	IIc	1) Aging degradation evaluation inadequate 2) Qualified life not established	Required documentation on file establishing qualification for aging. Aging analysis summary in EQR Section IX C.
64	GE Pressure Transmitter (Qty. 2)	Ib	1) Documentation Insufficient	Documentation establishing qualification not available. Replacement with qualified components scheduled during the next refueling outage. See EQR Action Item 6 for description and Justification for Continued Operation.

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POSITION INDICATION DEVICES

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
72	NAMCO Position Switch (Qty. 2)	Ia	None	Equipment Qualified
70, 71	NAMCO Position Switch (Qty. 13)	IIa	1) Similiarity of equipment and test specimen	Site inspection has verified that this equipment has a qualified sealing method which establishes similiarity between test specimen and installed equipment. See EQR Section VII F for description.
73, 79, 80, 81	NAMCO Position Switch (Qty. 10)	IIa	1) Documentation Insufficient	An analysis of functions has led to a reclassification of Exempt from qualification. Component not required for primary accident monitoring. See EQR Section VII O for evaluation.
74, 75, 78, 82, 83, 84, 85, 86	NAMCO, Microswitch Position Switch (Qty. 25)	IIa	1) Documentation Insufficient	Documentation establishing t from qualification not available. Replacement with qualified components scheduled during the next refueling outage. See EQR Action Item 33 for description and Justification for Continued Operation.
76, 77	Limitorque Position Transmitter (Qty. 2)	IIa	1) Documentation Insufficient	Required documentation in file establishing qualification. Document summary in EQR Section VII G.

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AUXILIARY COMPONENTS

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
69	GE Electric Heater (Qty. 4)	IIa	1) Documentation Insufficient	Component in an area requiring radiation qualification only. An analysis of subcomponent materials for radiation susceptibility establishes qualification. See EQR Section VII E for description.
87	Industrial Eng. Heaters (Qty. 2)	Ib	1) Documentation Insufficient	Equipment in an area harsh for radiation only. An analysis for thermal aging and radiation establishes qualification. See EQR Section II Action Item 23 for description.
108, 109, 112, 114	Okonite, Raychem Rockbestos, Anaconda instrument, Control and Power Cable	IIa	1) Similarity of test specimen and equipment inadequate 2) Aging degradation evaluation inadequate	Similarity analysis establishing qualification in EQR Section VII K. Aging analysis summary establishing qualification in EQR Section VII K.
111	Rockbestos Coaxial Cable	IIa	1) Testing failure criteria unsatisfactory 2) Functional testing unsatisfactory	Material identified incorrectly in submittal. Correct documentation establishing qualification in file. See EQR Section VII K for description.
110, 113	BIW, Rockbestos Coaxial Cable	IIc	1) Aging analysis inadequate 2) Qualified life not established	Aging analysis summary establishing qualification and qualified life in EQR Section IX D and IX E
107	Kerite Power Cable	IV	Documentation referenced not made available	Required documentation on file establishing qualification. See EQR Section VIII N for summary.

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AUXILIARY COMPONENTS

TER Equip. Item No.	Description	NRC Category	Deficiency	*Resolution Establishing Qualification
115	Victoreen instrument cable assembly	IIa	Similarity of test specimen and equipment inadequate	Required documentation on file establishing similarity. See EQR Section VII K for summary establishing qualification.
116, 117	Raychem cable splice Amerace terminal strips	Ia	None	Equipment qualified
118	General Electric Electrical Penetrations	IIa	1) Similarity of test specimen and equipment inadequate	Documentation establishing similarity on file. Description in EQR Section VII J.
			2) Aging degradation evaluation inadequate	Aging analysis establishing qualification summarized in EQR Section VII J.
			3) Spray criterion inadequate	Spray evaluation establishing qualification in EQR Section II Action item 20 description.

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ATTACHMENT 2

Accident Description, Environmental Conditions
and Equipment Identification

METHODOLOGY SUMMARY

"Definition of Postulated Accident Conditions

NRC IE Bulletin 79-01B defines the postulated accident conditions to be addressed in response to IE Bulletin 79-01B as loss-of-coolant accident/high-energy line break (LOCA/HELB) inside containment and HELB outside containment.

For the Duane Arnold Energy Center (DAEC), LOCA is analyzed in FSAR Section 14 and the high-energy pipe break analysis (both inside and outside containment) is discussed in FSAR Section 14.6 and the response to Questions and Responses 5.4 (Amendments 8 and 12).

For a HELB inside the drywell, worst-case environmental conditions are established by the postulated LOCA resulting from a recirculation line break as analyzed in FSAR Section 14.6.

For a HELB outside the drywell, the response to Q&R 5.4 submitted as part of Amendment 12 identifies high-energy lines and the location of postulated breaks. During analytical work performed in support of Amendment 12, environmental conditions were established.

Based on the above, the following HELB established the required post-accident environmental conditions for the DAEC:

A. LOCA inside the drywell

B. Main steam line break (MSLB) in the steam tunnel

C. Reactor core isolation cooling (RCIC) steam line break

D. High-pressure coolant injection (HPCI) steam line break

E. Reactor water cleanup (RWCU) break

F. Feedwater line break

Where components, due to their locations, could be subjected to differing environments for the various accidents, the most severe environmental conditions were utilized for analysis. Components located within compartments with postulated HELBs would be subjected to pressure, temperature, and humidity conditions resulting from the HELB, but would not be subjected to LOCA radiation doses simultaneously. These same components could be subjected to radiation doses following a LOCA, but these doses would not be simultaneous with pressure, temperature, and humidity. Components are evaluated to ensure that they are qualified for the environments in which they must operate.

Chemical spray and submergence outside the primary containment were not specifically analyzed by the FSAR, therefore an evaluation was done and conservation environmental conditions established. A summary of this evaluation is included below.

Chemical Spray

The DAEC design does not include a chemical spray accident. The spray system inside the drywell is limited to use of suppression pool water. For components located inside primary containment, the appropriate qualification parameters for water spray are reflected on the component evaluation sheet and each component evaluated to determine the effect of water spray on its qualification.

Submergence

Review of the DCR Guidelines for evaluating environmental qualification of Class 1E electrical equipment in operating reactor indicates that addressing submergence of safety-related electrical equipment outside containment is not required. However, an analysis has been performed in response to the SER request to address this subject. Calculations supporting this analysis were performed and are on file. The results of this analysis indicate that flood levels as a result of a HELB in the following rooms will not exceed the following values:

1. HPCI room: 5 inches
2. RCIC room: 1 inch
3. Torus room: 1 inch
4. Reactor water cleanup heat exchanger room: 7 inches
5. Reactor water cleanup pump room: 3 inches
6. Steam tunnel: 1 inch
7. Steam tunnel: 1 inch

For the turbine building condenser bay and feedwater pump area, there will be no appreciable accumulation of water due to the large expanse of unconfined area.

A review was made of the safety-related electrical equipment which is required to function to mitigate the consequences of the HELB in these areas. This review establishes that all the required equipment is located above the maximum flood elevation and therefore the equipment qualification parameters are not affected.

DESCRIPTION OF METHODOLOGY USED TO IDENTIFY EQUIPMENT WITHIN THE SCOPE OF 10 CFR 50.49 FOR THE ENVIRONMENTAL QUALIFICATION PROGRAM AT THE DAEC

1.0 INTRODUCTION

The methodology for review for system-specific failure effects is described in Section 2.0, the relevant plant design criteria are described in Section 3.0, and the results of including or excluding equipment from environmental qualification requirements based on Items a and b below are summarized in Section 4.0.

10 CFR 50.49(b)(2) includes in its scope nonsafety-related electric equipment whose failure under postulated harsh environment conditions could prevent satisfactory accomplishment of a safety function. The possibility of failure of nonsafety-related equipment in a manner detrimental to safety was evaluated previously by a combination of the following methods.

a. Review for system-specific failure effects undertaken during development of the DAEC response to NRC IE Bulletin 79-01B (and subsequent environmental qualification related submittals)

b. Review of plant design criteria which otherwise prevents (or limits) undesirable nonsafety-related equipment failure effects.

No distinction between safety-related and nonsafety-related equipment was made during development of the DAEC response to NRC IE Bulletin 79-01B.

2.0 REVIEW AND DEVELOPMENT OF THE LIST OF EQUIPMENT (SYSTEM COMPONENTS) REQUIRING ENVIRONMENTAL QUALIFICATION

The review conducted to identify equipment requiring environmental qualification was not limited to safety-related system components. Components required to operate to mitigate the consequences of a loss-of-coolant accident (LOCA) or high-energy line break (HELB) as defined in NRC IE Bulletin 79-01B or whose potential failure would affect plant safety were identified for review using the following methodology.

2.1 Equipment essential safety functions were determined based on Section 3 of the DOR Guidelines. These functions are as follows.

a. Achieve reactor cold shutdown

b. Achieve containment isolation

c. Provide reactor core cooling

d. Provide containment heat removal

e. Prevent the release of radioactive material in excess of the guidelines of 10 CFR 100.

2.2 Plant systems were reviewed to identify those essential for (or potential for a failure affecting) the safety functions presented in Section 2.1.

2.3 Piping and Instrumentation Diagrams for each of the systems identified in Section 2.2 were reviewed to identify system components essential to the performance of the system's safety function or whose failure would affect performance of the system's safety function and are required to function to mitigate the consequences of a LOCA or HELB as defined in NRC IE Bulletin 79-01B.

2.4 The entire instrument loop associated with each instrument identified in Section 2.3 was reviewed to identify other components whose function was essential or whose failure could adversely affect operation of the instrument loop.

2.5 Emergency Operating Procedures were reviewed to identify any additional equipment or instruments required to mitigate the postulated accidents.

2.6 The resultant component list was then coordinated with the NSSS vendor (General Electric) for completeness.

2.7 The environmental qualification equipment list submitted with the original DAEC response, dated Octiber 31, 1980, to NRC IE Bulletin 79-01B has been revised and upgraded on a continuing basis to reflect plant design changes and to incorporate accident monitoring instrumentation.

3.0 RELEVANT DAEC PLANT DESIGN CRITERIA

Plant design criteria preventing or limiting failure effects in nonsafety-related electric equipment are summarized below.

3.1 The DAEC plant design is in compliance with IEEE Standard 279-1971. Therefore, nonsafety-related equipment connected to protection systems (as defined in IEEE Standard 279-1971) are separated by approved isolation devices. No credible failure at the output of an isolation device will prevent the associated protection system channel from meeting the minimum performance requirements specified in the plant's design bases.

3.2 If the LOCA initiation signal occurs, nonessential electric loads are deenergized or transferred to the startup transformer (offsite power supply system). Therefore, failure of nonsafety-related electric equipment is prevented from affecting safety-related electric equipment.

3.3 Coordinated electrical fault protection (in the form of protective relays, circuit breakers, and fuses) is included in the plant design for electrical power distribution circuits. Therefore, harsh environment induced failures producing high current faults in nonsafety-related equipment will be prevented from affecting other electric loads.

3.4 Although not required (because of the above described plant design criteria) to address the 10 CFR 50.49 Paragraph b(2) nonsafety-related equipment failure concern, an additional level of confidence is provided by procurement specification requirements on cable used in both safety-related and nonsafety-related applications. In general (except for lighting cables). Iowa Electric-procured electric cable is environmentally qualified (including cable associated with nonsafety-related equipment). Therefore, harsh environment-induced failures in cables associated with nonsafety-related equipment (producing undesirable operation or failure of nonsafety-related equipment) will not occur.

4.0 RESULTS

The combination of the approach outlined in Section 2.0 and the plant design criteria of Section 3.0 resulted in the following inclusions and exclusions from environmental qualification requirements.

4.1 Electrical equipment which have been included in the DAEC environmental qualification program is summarized below.

a. Equipment whose active function is essential to the performance of a safety function (during a LOCA or other HELB)

b. Containment isolation valves and pilot solenoid valves in the case of air-operated control valves (regardless of whether the equipment is normally closed or fails safe)

c. Isolation valves which interface between nonsafety and safety systems

d. Isolation valve logic components

e. Accident monitoring instrumentation.

4.2 Electrical equipment excluded from the DAEC environmental qualification program is described below.

a. Instrument loops performing a nonessential alarm or display function

b. Nonessential electric loads tripped or transferred from essential buses by a LOCA initiation signal

c. Nonessential electric loads separated by isolation devices

d. Nonessential fail-safe devices normally in a fail-safe position where controls are located in a mild environment and no active safety function is required (excluding containment isolation valves, containment isolation pilot valves, and safety system interface isolation valves.).

e. Other electrical equipment not needed for mitigation of a LOCA or HELB.

4.3 CONCLUSIONS

As described in Section 2.0, the scope of the system review effort included both safety-related and nonsafety-related equipment. On a system and instrument loop basis, potential failures affecting performance of system safety functions were investigated. Although nonsafety-related equipment within the scope of Paragraph b(2) of 10 CFR 50.49 has not been uniquely identified in environmental qualification submittals to the NRC, equipment with potential for a failure (in a manner detrimental to safety) have been included in the DAEC environmental qualification program.

ACCIDENT MONITORING INSTRUMENTATION

The first and second semiannual environmental qualification reports provided a list of accident monitoring instruments and associated environmental qualification summary sheets based on a review of the existing DAEC emergency procedures. Accident monitoring instrumentation continues to be the subject of other Iowa Electric licensing activities (as described above and elsewhere). For purposes of this report, Regulatory Guide 1.97, Revision 2 and the associated BWR Owners Group position were used as guidance in establishing (to the extent possible) the interim list of DAEC accident monitoring instrumentation requiring environmental qualification.

ATTACHMENT 3

Equipment Qualification Report
Sections II, VI, VII, VIII, IX

II. ACTION PLAN

The information provided in this section is an item-by-item description of intended Iowa Electric action to correct or resolve cases of incomplete documentation supporting environmental qualification. In each case, a description of the action item, its resolution status, and JCO are provided. Where action is complete, a reference to the most recent semiannual update report describing the action item and its final resolution is made.

NOTE: The JCOs submitted with Reference 4 (i.e., for equipment within the scope of Action Items 6, 10, 12, 14, 17, 18, 19, 20, 22, 23, 24, 25, 27, 28, 29, and 30) were reviewed and found technically acceptable as stated in Appendix D of TER-C5257-499 (for the DAEC), dated August 18, 1982.

In this section, the applicable regulatory document (to which environmental qualification compliance is being sought) is indicated where resolution of an action item requires equipment replacement. In general, compliance with NUREG 0588, Category I is indicated for cases where a purchase order citing such requirements has already been issued. For the remaining cases, compliance with 10 CFR 50.49 (which is understood to supersede NUREG 0588) is indicated. Additional DAEC replacement equipment philosophy is provided in Section III.D.

1. ASCo SOLENOID VALVES

   Action complete (see Reference 2).

2. MAIN STEAM SAFETY RELIEF SOLENOID VALVES

   Action complete (see Reference 2).

3. TERMINAL BLOCKS

   Action complete (see Reference 2).

4. SOLENOID VALVE SV-4310

   Action complete (see Reference 3).

5. MAIN STEAM ISOLATION VALVE POSITION SWITCHES

   Action complete (see Reference 2).

6. PRESSURE TRANSMITTERS PDT-2046, -1947

a. Action Item Description

These pressure transmitters sense differential pressure between the tubes and shell of the RHR heat exchanger and function to maintain service water pressure greater than RHR system pressure to prevent radioactive leakage into RHR service water. These transmitters must be qualified for a radiation dose of 5.9×10^6 rads. Documentation demonstrating qualification is not available.

b. Resolution

These components will be replaced during the Cycle 8 refueling outage with transmitters qualified to NUREG 0588, Category 1.

c. Justification for Continued Operation

Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. During this time, action will be taken to shut down redundant systems beyond those needed for coolant injection. This will result in radiation exposures to only one of two redundant equipment trains. Also, actual doses will be less than those calculated using such conservative assumptions. Finally, if the operating equipment suffers degradation (resulting in failure) due to radiation exposure, the failure will be detected by the operators via system alarms; the operators will then activate the redundant system train. During time available after stabilization of core cooling, additional coolant injection paths will be established, if necessary, to ensure long-term cooling.

7. RHR PUMP MOTORS 1P-229A,B,C,D

Action complete (see Reference 4)

8. CORE SPRAY PUMP MOTORS 1P-211A,B

Action complete (see Reference 4)

9. LEVEL SWITCHES LS-1861A,B,C,D

Action complete (see Reference 4)

10. FLOWMETERS FM-8408A,B,C,D

a. Action Item Description

These flowmeters monitor main steam line leakage flow. They are S.K. Instrument Model 20-9651-8550, but were supplied by General Electric. These flowmeters provided a signal to their respective flow switches which cause system isolation on high flow. The flow switches are located in the control room (mild environment). The flowmeters and associated transmitter circuitry are located in the steam tunnel and must be qualified for a total dose of 2.1×10^7 rads. Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required. S.K. Instrument has qualified the flowmeter for 1×10^6 rads. Additional qualification documentation is required.

b. Resolution

New flow sensors qualified to NUREG 0588, Category I will be installed outside the steam tunnel to provide the signal to perform the safety function of system isolation on high flow. The sensors are being procured from Fluid Components, Inc. and will be installed during the Cycle 8 refueling outage via DCR 1095.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons:

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) Failure of these components could, at worst, cause the failure of the main steam isolation valve (MSIV) leakage control system to function properly. In this case, radioactive leakage past the MSIVs, which is expected to be minimal, will normally be contained by the main steam piping outside the drywell.

11. HEATERS 1S-122A,B,C,D

Action complete (see Reference 3).

12. EXHAUST UNITS 1K-25A,B

Action complete (see Reference 5).

13. MOTOR CONTROL CENTER 1D41

Action complete (see Reference 3).

14. AIR COOLING UNITS 1V-AC-11,12

a. Action Item Description

The motors for these air cooling units are Westinghouse with TEFC type enclosures. These units function to provide post-accident room cooling for the RHR rooms. These units are located in the RHR rooms and must be qualified for a total dose of 5.9×10^6 rads. Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required.

b. Resolution

Subsequent investigation and evaluation indicate that it is not feasible to document qualification of these motors. They will be replaced during the Cycle 8 refueling outage with new motors qualified to NUREG 0588, Category I. Installation will be via DCR 1148.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models.

Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) Failure of the room cooling unit would, at worst, result in elevated room temperatures during emergency conditions. A study was recently completed to evaluate the effect of loss of room cooling in the HPCI room. The results of this study, which are conservatively analogous to the RHR corner rooms, show a temperature rise of approximately 13F in 2 hours using conservative assumptions. Under realistic room and environmental conditions, it is expected that the room temperature will remain near or below the maximum design room temperature.

3) Although each RHR corner room contains one cooling unit, the corner rooms and associated safety-related equipment provide redundant safety system capability. Because of operational considerations, both redundant safety systems will not be required to operate continuously. Therefore, radiation exposure (from process fluid) to the cooling fan motors associated with each train will not be the same, resulting in longer overall RHR corner room temperature control capability. Also, in the longer term, corner room process fluid heat load will be reduced.

15. AIR COOLING UNITS 1V-AC-14A,B

Action complete (see Reference 4)

16. MOTOR-OPERATED VALVES

Action complete (see Reference 3).

17. FLOW SWITCHES FIS-2111, 2131

Action complete (see Reference 5).

18. PRESSURE SWITCHES PDIS-1971A,B

Action complete (see Reference 5).

19. LIMITORQUE VALVE OPERATOR MOTOR BRAKES (MANUFACTURER AND MODEL TYPE VARIOUS)

a. Action Item Description

Environmental qualification documentation is not available to support a post-accident function capability of motor brakes contained in Limitorque valve operators. Limitorque valve operator records have been reviewed to confirm the list of harsh environment valve operators containing motor brakes. The following motor operators have been determined to contain motor brakes: MO-1909, MO-1908, MO-2238, MO-2239, MO-1902, MO-2135, MO-2115, MO-2117, MO-2137, MO-2003, MO-2000, MO-1905, MO-4627, and MO-4628.

MO-2000, MO-2115, MO-2135, and MO-2137 are located in areas of the reactor building that are harsh for radiation only and are required to be qualified for a total radiation dose of 7.5×10^5 rads. The remaining motor operators are in locations subject to LOCA or other HELB effects in addition to post-LOCA radiation doses. These motor operators must be qualified for a total radiation dose in the range of 7.5×10^5 rads to 4.3×10^7 rads (depending on location). Environmental qualification of these motor brakes is under investigation.

b. Resolution

An investigation (which has included walkdowns for nameplate information and communications with valve operator, motor, and motor brake manufacturers) has provided the following motor brake summary information.

Plant ID	Brake Manufacturer/Model	Location
MO-1908	Dings/R71010-7	Drywell
MO-2238	Dings/R71015-7	Drywell
MO-4627	Dings/2-63015-24	Drywell
MO-4628	Dings/2-63015-24	Drywell
MO-1905	Dings/6-83075-19	RHR valve room
MO-2003	Dings/6-83075-19	RHR valve room
MO-2117	Dings/X6-71015-29	RWCU heat exchanger room

Plant ID	Brake Manufacturer/Model	Location
MO-1902	Dings/X6-71010-29	RHR valve room
MO-2000	Dings/X6-71010-29	RB south
MO-2137	Dings/X6-71015-29	RB north
MO-2135	Dings/X6-71015-29	RB north
MO-2115	Dings/X6-71015-29	RB south
MO-1909	Stearns/1-087-035	RHR valve room
MO-2239	Stearns/1-087-035	Steam tunnel

Dings Models R71010-7 and R71015-7 are similar in principle of operation, construction, and subcomponent materials to Dings Model 6-61009-50 tested by FRC (Test Report F-C2232-01, dated November 1, 1968) for Limitorque Corporation with the exception of a gasket between the brake enclosure housing and brake bracket which functions to ensure that the brake is waterproof. Installation in Models R71010-7 and R71015-7 of a housing gasket material qualified for the post-LOCA environmental service conditions (for construction consistency with Spec 50 models) will be accomplished before or during the Cycle 8 refueling outage.

NOTE: Each "Spec 50" Dings motor brake model (i.e., model number of Form X-XXXXX-50) can also be shown to be similar in principle of operation, construction, and subcomponent materials. Using Test Report F-C2232-01 as a basis and subcomponent materials information available from Dings Models R71010-7 and R71015-7 (and the Spec 50 models) have been shown to meet the criteria of the Division of Operating Reactors (DOR) guidelines for their DAEC application.

Environmental qualification of the motor brakes for MO-2000, MO-2115, MO-2135, and MO-2137 has been established by engineering analysis of the radiation and thermal aging effects of the motor brake's subcomponent materials. This analysis was completed in August 1983.

The remaining motor brakes will be replaced with the Spec 50 Dings model (which is qualified to the DOR Guidelines) during or before the Cycle 8 refueling outage. This intended resolution is understood to be consistent with NRC requirements for spare and replacement parts (reference NRC Generic Letter 82-09 dated April 20, 1982) because sound reasons for the intended action exist as summarized below.

1) The Spec 50 Dings model represents the best motor brake available and it is qualified to the DOR Guidelines. Neither motor brakes qualified to NUREG 0588 nor incentive for the industry to provide a NUREG 0588 qualified motor brake exist because currently available motor-operated valves do not utilize motor brakes.

2) Because of the relatively simple principle of operation and construction (spring solenoid-actuated disk brake) and available subcomponent materials information, no safety advantage exists between a NUREG 0588 qualified model and a model which satisfies the DOR Guidelines.

3) The additional cost and radiation exposure to maintenance personnel incurred from replacing the valve actuator or modifying it (to allow deletion of the brake), as compared to replacing the motor brake, cannot be justified for safety reasons.

   If a compatible replacement motor brake (meeting minimum environmental qualification requirements) cannot be located, the valve operator will be modified to eliminate the need for the motor brake.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) As was demonstrated during Limitorque valve operator qualification tests, the weatherproof operator housing minimizes the effect of harsh environmental parameters except for radiation. Therefore, the primary concern of an unqualified motor brake would be radiation induced failure of the motor brake solenoid (resulting in locking the valve in its position at the time of failure). All valve operators identified above perform their safety function immediately upon detection of accident initiation and prior to being subjected to a radiation dose of a value likely to cause motor brake failure.

2) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases which occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

3) Additional justification is provided below.

   a) Valves MO-1905 and MO-2003 are containment isolation valves in the low-pressure coolant injection (LPCI) lines. These valves are normally closed and will remain so following an accident until it becomes necessary to initiate core cooling with the LPCI system. In the event that these valves fail to open at that time, this will cause a loss of LPCI capability. An analogous scenario has been evaluated in response to FSAR Question 9-6.4 on Page 9-6.4-1 of the original DAEC FSAR. The conclusion for case three of that question wherein "no LPCI flow will enter the vessel" is that it "will not result in a peak cladding temperature greater than that presented in the FSAR." Therefore, the consequences of the potential failure of these valves has been evaluated and found acceptable.

   b) Valves MO-2238 and MO-2239 are containment isolation valves in the HPCI system. As discussed in Section III.A of this report, the HPCI system need not be environmentally qualified for the HPCI function. Should these valves be required to perform their containment isolation function, the HPCI function will no longer be required and containment isolation will occur prior to the valves receiving a significant radiation dose. Following closure for containment isolation, the valves need not reopen.

c) Valve MO-1902 is a containment isolation valve for one of the containment atmosphere spray headers in the RHR system. This valve is normally closed and will remain so unless the operator elects to manually initiate drywell or suppression pool spray. If the valve fails to open, the operator will be denied containment spray capability. In evaluating the design basis accident (DBA), the DAEC FSAR does not take credit for operation of the containment spray system, therefore, inability to initiate containment spray will not impair the ability for safe shutdown following a DBA.

d) Valve MO-2117 is a core spray pump discharge isolation valve. This valve is normally closed and must open to provide core spray flow to the reactor vessel. The environment surrounding this valve does not become harsh until after the valve performs its safety function. The radiation dose to the valve operator is primarily a result of the radioactive process fluid flowing through the valve after it opens.

e) Valves MO-4627 and MO-4628 are recirculation pump discharge isolation valves (one for each loop). These valves are normally open but close in the event of high drywell pressure or low reactor vessel level. This directs flow from the LPCI system through the recirculation loop directly to the reactor vessel. In the event of a failure of these valves to close, core cooling flow continues to be provided, although a portion of the flow will be through the suction side of the intact recirculation loop backwards through the recirculation pump and into the reactor vessel. Additional cooling is also provided by the redundant core spray system.

f) Valves MO-1908 and MO-1909 are containment isolation valves in the RHR supply line from the recirculation system. These valves are normally closed

and will remain so following a DBA. The shutdown cooling mode of the RHR system is not required following a DBA with its high radiation source terms; therefore, the valves remain in the closed position and will be unaffected by a brake failure.

20. GENERAL ELECTRIC ELECTRICAL PENETRATIONS CANISTER TYPE MODELS NS02-I, NS02-II, NS03, AND NS04

a. Action Item Description

The following electrical penetrations (General Electric canister type Models NS02-I, NS02-II, NS03, and NS04) provide cable penetrations into the drywell.

JX-100A	JX-100B	JX-100C	JX-100D
JX-103	JX-105A	JX-105B	JX-105C
JX-105D	JX-101A	JX-101B	JX-104A
JX-104B	JX-104C	JX-104D	

Note: Electrical penetrations JX-100A, B, C, and D have been included within the scope of this action item consistent with the electrical support function which they provide for the post-TMI-2 modification to add containment high-range radiation monitors (DCR 909).

The penetrations are required to be qualified for LOCA conditions inside drywell. The penetrations were not qualified for spray by test. Additional investigation into post-LOCA operability during demineralized water spray conditions is required.

A subcomponent analysis being conducted on the penetration assemblies has also determined that penetrations JX-105A through JX-105D and JX-104A may utilize nylon-insulated splices. Although this splice material is qualified for post-LOCA drywell conditions, additional evaluation of susceptibility to aging degradation is required. For the limiting type of nylon, the potential for significant aging degradation (affecting post-accident operability) will not occur prior to 10 years of operation (reference DOR Guidelines, Table C-1) (i.e., approximately 1984 for the DAEC). Investigation for identification of the type of nylon is required to allow further determination and refinement of qualified life (beyond 10 years) and associated surveillance/modification requirements.

b. Resolution

The electrical penetrations are shielded from the effects of water spray by a metal enclosure that totally surrounds the penetration and electrical termination area. The enclosure's orientation and adequacy for shielding against water spray was evaluated to be adequate as a result of a walkdown during the Cycle 7 refueling outage.

Resolution of the splice qualification concern will be accomplished by replacement or modification of the existing splices before or during the Cycle 8 refueling outage.

c. Justification for Continued Operation

Continued station operation is justified because the subject penetrations have been tested and environmentally qualified for severe LOCA environmental conditions, including high humidity and superheated steam. The penetration assemblies are inherently designed to prevent the intrusion of moisture into critical components. Because of its protection from direct spray effects, the splice materials do not require continued mechanical strength to perform its safety function of electrical insulation during a LOCA. When not in tension or under continued stress, nylon's ability to withstand aging and radiation dose is increased. Also, potential for significant aging degradation of nylon will not occur until after at least 10 years of operation (reference DOR Guidelines, Table C-1). Therefore, design basis LOCA conditions are not expected to impact the environmental capability of the penetrations.

21. AUTOMATIC VALVE COMPANY SOLENOID VALVE MODEL C5450-5

Action complete (see Reference 4)

22. FENWAL CONTROL UNITS MODEL 35003-0

a. Action Item Description

Control units CU-5835A1, A2, B1, B2, and CU-5837A1, A2, B1, B2 (Fenwal Model 35003-0) are located in the standby gas treatment system (SGTS) room at elevation 786' in the reactor building. They are required to function as part of the SGTS. The

control units must be qualified for a 30-day integrated radiation dose of 1.6×10^8 rads. Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required. Each control unit consists of a metallic sensor and an electronic switch. An analysis of radiation and thermal aging effects of the sensor, including its extension cable has determined its acceptability for at least 4.8×10^8 rads and a 40-year qualified life. Qualification documentation is not available for the electronic switch.

b. Resolution

The electronic switches were relocated to a mild environment during the Cycle 7 refueling outage. Relocation was accomplished via DCR 1121.

23. ELECTRODYNE VALVE OPERATOR MODEL TN-24-400

a. Action Item Description

Valve operators MO-1904 and MO-2004 (Electrodyne Model TN-24-400) are located in the RHR valve room and are required to function as part of the RHR. These operators are required to be qualified for a temperature of 277F, a pressure of 1.2 psig, a relative humidity of 100%, and a 40-year normal plus 30-day accident integrated radiation dose of 5.6×10^6 rads.

b. Resolution

The DAEC valve operator model (with the exception of the motor and motor brake) has been determined to be similar in construction and operating principle to the Electrodyne model (TN200) tested by Franklin Research Institute Laboratory in Test Report F-C2883, dated March 1971, for Link-Belt Division of FMC Corporation. The environmental qualification concern is limited to the motor (Allis-Chalmers Model 012) and motor brake (Stearns Electric Model 1-081-011) of each of the valve operators.

Replacement motor and motor brake combinations acceptable with respect to both compatibility (with the existing valve operators) and environmental qualification have not been located. Therefore,

unless an acceptable motor and motor brake combination is located, the entire valve operator assemblies will be replaced with an equivalent model manufactured by Limitorque Corporation. This replacement operator will be qualified in accordance with 10 CFR 50.49 to the extent possible. If the replacement model requires a motor brake, the new brake will be qualified consistent with NRC policy for replacement parts (see resolution of Action Item 19).

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) The room in which the valves are located does not contain a high-energy line; however, it does communicate (via an unsealed pipe chase) with the torus room in which a HELB is postulated. Therefore, the HELB conditions in the torus room have been conservatively applied to the room containing these valves. Because of the remoteness of these valves from the source of the HELB, the actual environmental conditions at the valve's location will be less than specified.

24. ASCo SOLENOID VALVES MODELS HT831665, 831665, AND 8320A6; HB8302C25RU

a. Action Item Description

The following solenoid valves (ASCo Models HT831665, 831665, 8320A6, and HB8302C25RU) are located in various areas at the DAEC and are required to perform safety functions in several different systems.

SV-1963	SV-1966	SV-2033	SV-2037
SV-5815A	SV-5815B	SV-5825A	SV-5825B
SV-5801A	SV-5801B	SV-7602A	SV-7602B
SV-4303	SV-4306	SV-4307	SV-4308
SV-4311	SV-4312	SV-4313	SV-4640
			SV-4309

These solenoid valves are located in areas that are harsh for radiation only, with the required doses ranging from 2.9×10^5 rads to 1.6×10^8 rads.

b. Resolution

These valves were replaced during the Cycle 7 refueling outage via DCR 1109. The Model 831665 and HT831665 valves were replaced by ASCo Model NP831665E; the Model 8320A6 was replaced by ASCo Model NP8320A173E; and the Model HB8302C25RU was replaced by ASCo Model 206-832-2U. All these replacement models are qualified for a radiation dose in excess of 1.6×10^8 rads and to the requirements of NUREG 0588, Category I.

25. PENN TEMPERATURE SWITCH MODEL A-19ABB-6

a. Action Item Description

Temperature switches TS-5808A and TS-5808B (Penn Model A-19ABB-6) are located in the SGTS room at elevation 786' in the reactor building. They are required to function as part of the SGTS. Their specific location in the SGTS requires qualification for a design basis total radiation dose of 2.5×10^7 rads. This required radiation dose accounts for distance attenuation from the primary radiation service within the room. Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required.

b. Resolution

Qualification by analysis is not possible because of insufficient radiation qualification information associated with the temperature switch bulb fill fluid. The existing temperature switches will be replaced during the Cycle 8 refueling outage with temperature switches (or a temperature element and switch combination) qualified to 10 CFR 50.49 requirements. Installation will be via DCR 1142.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) At the start of an accident, both trains of the SGTS will be automatically started. Plant operating procedures require that one train be manually isolated such that only one train is operated at a time. The dominant radiation source for the SGTS is the loading of the SGTS filters. Therefore, the train which is isolated initially following an accident will not experience the same radiation doses as the operating train. Should the initially operated train fail, the redundant train can be restarted to maintain the SGTS function.

26. GULTON INDUSTRIES COMPANY TEMPERATURE ELEMENT MODEL TCA-0646

Action complete (see Reference 4)

27. ESSEX CONTROLS TEMPERATURE SWITCHES (MODELS 351-34912 AND 351-253924)

Action complete (see Reference 5).

28. INDUSTRIAL ENGINEERING EQUIPMENT COMPANY HEATERS MODEL TFZCP15900

Action complete (see Reference 5).

29. ROSEMOUNT TEMPERATURE ELEMENTS MODEL 104MA23ABBB

a. Action Item Description

The following temperature elements are located in the steam tunnel and turbine building near the main steam lines and are required to function for main steam line break (MSLB) leakage detection.

TE-4443A	TE-4443B	TE-4443C	TE-4443D
TE-4444A	TE-4444B	TE-4444C	TE-4444C
TE-4445A	TE-4445B	TE-4445C	TE-4445D
TE-4446A	TE-4446B	TE-4446C	TE-4446D
TE-4477A	TE-4477B	TE-4478A	TE-4478B
TE-4479A	TE-4479B	TE-4480A	TE-4480B

These temperature elements are required to be qualified for a temperature of 300F, a pressure of 1.8 psig, a relative humidity of 100%, and a 40-year integrated dose of 7.2×10^6 rads.

b. Resolution

These temperature elements will be replaced with new Pyco temperature elements meeting NUREG 0588, Category I requirements. Replacement, scheduled for the Cycle 8 refueling outage, will be via DCR 1161.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) The equipment qualification concern is the quantitative effect of radiation on the mechanical and electrical properties of each temperature element's teflon-insulated lead wire. This lead wire is contained inside a weatherproof head and is not subject to mechanical stress. Also, it is not essential that electrical insulation maintain its mechanical strength and other properties for proper operation of the temperature element. With respect to electrical properties, only dielectric strength and electrical resistance are important but these properties are not permanently affected by radiation dose. Although teflon's electrical resistance and dielectric strength are somewhat affected by radiation dose rate, these temperature elements do not perform a safety function during accidents that produce high radiation dose rates. Therefore, it is unlikely that degradation of the insulation due to radiation damage will result in failure of the instrument.

3) These temperature elements function to close the MSIVs in the event of a MSLB. Because of redundancy, all temperature elements in a given area would be required to fail to prevent the main steam line isolation for a steam line break in that area.

4) For design-basis MSLBs, diverse means of detecting the accident (such as reactor vessel low water level or main steam line high flow) exist and will result in automatic closure of the main steam isolation valves.

30. BARKSDALE PRESSURE SWITCHES MODEL P1H-M85SS-V

Action complete (see Reference 5).

31. TEMPERATURE ELEMENTS FOR SUPPRESSION POOL WATER TEMPERATURE MONITORING (BURNS TYPE E)

a. Action Item Description

Suppression pool water temperature is currently being monitored by temperature elements TE-4324 and TE-4325. They are Burns Type E RTD temperature elements. These suppression pool water temperature elements must be qualified for a total radiation dose of 3.5×10^7 rads (40-year normal plus 30-day LOCA) and postulated HELBs in the torus room (peak temperature 277F, peak pressure 1.2 psig, and 100% humidity). Documentation demonstrating qualification of TE-4325 and TE-4324 is not available.

b. Resolution

Resolution of the qualification of temperature elements TE-4324 and TE-4325 will be addressed as part of the review of NUREG 0737, Supplement 1, as described in Section IV of this report.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) This equipment performs no automatic safety function; therefore, failure of this equipment will not result in failure of an automatic safety system to perform its safety function.

3) The accident conditions producing harsh effects at the RTD location are either design basis LOCA or a HELB in the torus room.

   a) Design basis LOCA: The equipment qualification concern is limited to the post-accident radiation dose. The subcomponents that would be susceptible to radiation degradation are the terminal blocks and lead wire insulation. If the lead wire and terminal block experiences radiation degradation, failure of the RTD to provide its temperature dependent signal is unlikely because neither the lead wire nor the terminal block are subject to mechanical stress of a level sufficient to result in gross subcomponent failure. Also, the most important use of information provided by these RTDs would be in the early stages of a LOCA when rapid suppression pool heatup occurs and radiation dose levels are less than long-term post-accident.

   b) HELB postulated in the torus room [i.e., potential break in either the HPCI system steam supply or reactor core isolation cooling (RCIC) system steam supply piping]: For such postulated accidents, useful information is provided by these RTDs only in the event that safety relief valves lift (with resultant discharge to

the suppression pool). In such a case, alternative and diverse means of monitoring torus conditions is available (such as torus water level indication and RHR heat exchanger direct and recorded temperature indication).

4. RTDs are relatively uncomplicated temperature monitoring devices and each manufacturer model type is of similar construction and principle of operation. Because several qualified models exist (e.g., Pyco and Conax), additional reliability exists as to harsh environment post-accident operability.

5. Failure of these RTDs (to provide a representative temperature-dependent signal) is unlikely as described above and will not result in the operators taking action affecting plant safety because both redundant and diverse means of determining instrument failure exist. (Redundant: because of nonuniformity in actual harsh environment conditions at the two RTD locations, both RTDs are unlikely to fail simultaneously; Diverse: alternative indications of torus conditions exist such as torus water level, safety relief valve temperature and discharge downstream pressure switch indication, and RHR heat exchanger inlet temperature.)

32. POSITION SWITCHES FOR PRIMARY CONTAINMENT ISOLATION VALVE POSITION INDICATION (MANUFACTURER AND MODEL NUMBER VARIOUS)

a. Action Item Description

The following position switches monitor position of containment isolation valves external to the drywell and are required to be qualified for post-accident radiation dose only. Adequate environmental qualification documentation is not available.

Plant Identification	Manufacturer/Model	Required Radiation Dose (rads)
ZS-3704	Microswitch/DTF2-2RN-RH	2.7 E06
ZS-3705	Microswitch/DTF2-2RN-RH	2.7 E06
ZS-3728	Microswitch/DTF2-2RN-RH	2.7 E06

Plant Identification	Manufacturer/Model	Required Radiation Dose (rads)
ZS-3729	Microswitch/DTF2-2RN-RH	2.7 E06
ZS-4304	Microswitch/OPD-AR	2.9 E05
ZS-4305	Microswitch/OPD-AR	2.9 E05
ZS-4306	Microswitch/OPD-AR	1.5 E06
ZS-4307	Microswitch/OPD-AR	1.5 E06
ZS-4308	Microswitch/OPD-AR	1.5 E06
ZS-4311	Microswitch/DTF2-2RN-RH	1.5 E06
ZS-4312	Microswitch/DTF2-2RN-RH	1.5 E06
ZS-4313	Microswitch/DTF2-2RN-RH	1.5 E06
ZS-4331A	Target Rock/72V-004	5.6 E06
ZS-4331B	Target Rock/72V-004	5.6 E06
ZS-4332A	Target Rock/72V-004	7.5 E05
ZS-4332B	Target Rock/72V-004	7.5 E05
ZS-4333A	Target Rock/72V-004	1.3 E07
ZS-4333B	Target Rock/72V-004	1.3 E07
ZS-4334A	Target Rock/72V-004	1.3 E07
ZS-4334B	Target Rock/72V-004	1.3 E07
ZS-4640	NAMCo/SAI-131	1.1 E06
ZS-5703A	Microswitch/DTF2-2RN-RH	2.7 E06*
ZS-5703B	Microswitch/DTF2-2RN-RH	2.7 E06*
ZS-5704A	Microswitch/DTF2-2RN-RH	1.3 E07
ZS-5704B	Microswitch/DTF2-2RN-RH	1.3 E07
ZS-5718A	Microswitch/DTF2-2RN-RH	1.3 E07
ZS-5718B	Microswitch/DTF2-2RN-RH	1.3 E07
ZS-5719A	Microswitch/DTF2-2RN-RH	2.7 E06*
ZS-5719B	Microswitch/DTF2-2RN-RH	2.7 E06*

*Dose was reduced as a result of reevaluation of post-accident operating time.

b. Resolution

Position switches ZS-4331A and ZS-4331B through ZS-4334A and ZS-4334B have been determined to be qualified in accordance with Target Rock Test Report 2302, dated May 9, 1979.

Position switches for ZS-4640, ZS-5704A and ZS-5704B, and ZS-5718A and ZS-5718B were confirmed to not be qualifiable by engineering analysis and will be replaced with position switches qualified in accordance with 10 CFR 50.49 during or before the Cycle 8 refueling outage.

An evaluation of the radiation and thermal aging effects of the subcomponent materials of the remaining position switches was completed in June 1983. The results are as follows.

Position switches ZS-4304 through ZS-4308 are qualified for a total radiation dose of 3.0×10^6 rads and a thermal aging qualification of 30 years. Position switches ZS-3704, ZS-3705, ZS-3728, ZS-3729, ZS-4311 through ZS-4313, ZS-5703A and ZS-5703B, and ZS-5719A and ZS-5719B are qualified for a total radiation dose of 4.7×10^6 rads and a thermal aging qualification of 30 years.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) The above position switches perform no automatic safety function; therefore, failure of these position switches will not result in a failure of an automatic safety system to perform its safety function.

3) Failure of a valve position switch is unlikely to mislead an operator because realistic harsh environment-induced failure modes will be apparent. Potential failures include either short circuits resulting in both "open" and "closed" lights on or open circuits resulting in both lights off; it is unlikely that a failure can occur resulting in the exact opposite valve position indication.

4) With the exception of the drywell cooling water system (see 7.b below), isolation valves are redundant. Failure of both valves would be required for containment integrity to be affected.

5) Any radioactive releases would be detected by effluent radiation monitors. Operator action would then result in system isolation at the radioactive release point, thus maintaining containment integrity.

6) These position switches are outside of primary containment and are harsh for radiation-only environments. The materials used in the contacts of position switches (such as phenolics) are, in general, acceptable for at least 10^6 rads.

7) Failure of these position switches will not result in the operators taking action affecting plant safety because of the following backup indication or knowledge of system design, construction, and principles of operation.

   a) Position switch ZS-4640 provides control valve position indication for a reactor recirculation sample line isolation valve. This valve is normally closed and remains closed following a LOCA. Position indication of the upstream valve CV-4639 (also normally closed) is provided by ZS-4639 which is qualified for its postulated environmental conditions.

   b) Position switches ZS-5704A, ZS-5704B, ZS-5718A, and ZS-5718B provide control valve position indication for the drywell cooling water system isolation valves. The drywell cooling water system is a closed system and does not communicate directly with the drywell atmosphere or penetrate the reactor coolant system (RCS) pressure boundary. Also, the valves associated with position switches ZS-5718A and ZS-5718B are backed up by check valves which further prevent or reduce the likelihood of uncontrolled radioactivity releases through this system.

33. TEMPERATURE ELEMENTS FOR DRYWELL ATMOSPHERE TEMPERATURE INDICATION (LEEDS & NORTHRUP MODELS 8920-404-00-3-21 and 8197-10-S)

a. Action Item Description

Drywell atmosphere temperature is presently monitored by the following temperature elements (located in the drywell).

Model 8920-404-00-3-21		Model 8197-10-S	
TE-4386E	TE-4386F	TE-4328E	TE-4328F
TE-4386G	TE-4386H	TE-4328G	TE-4328H
TE-4386J	TE-4386K	TE-4328J	TE-4328K
TE-4386L	TE-4386M	TE-4328L	TE-4328M

The temperature elements are manufactured by Leeds & Northrup. They are RTD type temperature elements and must be qualified for a total radiation dose of 4.3×10^7 rads (40-year normal plus 30-day LOCA) and drywell LOCA conditions (peak temperature 324F, peak pressure 62 psig, and 100% humidity). Documentation of qualification is not available.

b. Resolution

An evaluation was completed in July 1983 to identify the number of temperature elements required to representatively monitor drywell temperature. This evaluation concluded that the eight TE-4386 series temperature elements are sufficiently representative in location to provide acceptable drywell post-accident temperature location. The TE-4328 series temperature elements are special purpose RTDs that are used during integrated leak rate testing.

The eight TE-4386 series temperature elements will be replaced with Pyco-manufactured temperature elements qualified in accordance with 10 CFR 50.49 during or before the Cycle 8 refueling outage via DCR 1161.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) This equipment performs no automatic safety function; therefore, failure of this equipment will not result in a failure of an automatic safety system to perform its safety function.

3) The temperature elements are located in various regions of the drywell such that the environment is different at each element. The probability of all elements failing is small. Because all elements are measuring temperature inside primary containment, the failure of any one element that would cause a grossly erroneous reading would be easily detected by comparison to other nearby elements and will not result in the operators taking action affecting plant safety.

4) For the accident (design basis LOCA) and the time frame of concern, saturation conditions will exist in the drywell such that post-accident drywell temperature can be approximated by use of existing qualified drywell pressure transmitters and steam table data.

34. PRESSURE TRANSMITTERS FOR MSIV LEAKAGE CONTROL SYSTEM PRESSURE INDICATION (GE MODEL 555111DEAA4WCB)

a. Action Item Description

The MSIV leakage control system (LCS) pressure is presently monitored by pressure transmitters PT-8404A through PT-8404D (which monitor pressure between the MSIVs). They are GE Model 555111DEAA4WCB and sense pressure by means of a stainless steel diaphragm. The transmitters are located above the control rod drive repair room and must be qualified for a total radiation dose of 8.9×10^5 rads. Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required. Documentation demonstrating qualification is not available.

b. Resolution

MSIV-LCS pressure provides information relative to system operation. A more direct indication of MSIV-LCS operation is provided by system flowrate. (Note: Pressure boundary indication will be addressed by a separate accident monitoring system upgrade.) This action item can be considered resolved after installation of NUREG 0588, Category I qualified flow sensors (see Action Item 10) during or before the Cycle 8 refueling outage via DCR 1095.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) Failure of these components will not cause the failure of the MSIV leakage control system to function properly.

3) Failure of these instruments will not result in the operators taking action affecting plant safety because alternative indications of proper MSIV leakage control system operation exist (such as system exhaust blower run indication, system valve position, and status lights associated with system permissive switches).

35. FLOW TRANSMITTERS FOR CORE SPRAY FLOW INDICATION (GE MODEL 555-111BDAA3PDF)

a. Action Item Description

Core spray flow is presently monitored by flow transmitters FT-2130 and FT-2110. These transmitters are GE Model 555-111BDAA3PDF and are located in the RHR corner rooms in panels 1C-123 (FT-2110) and 1C-124 (FT-2130). They must be qualified for a 40-year normal plus 30-day post-accident radiation dose of 2.8×10^6 rads. Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required. Documentation of qualification is not available.

b. Resolution

Flow transmitters FT-2130 and FT-2110 will be replaced with new flow transmitters qualified in accordance with 10 CFR 50.49 during or before the Cycle 8 refueling outage.

c. Justification for Continued Operation

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) These transmitters perform no automatic safety function; therefore, failure of these instruments will not result in a failure of an automatic safety system to perform its safety function.

3) Failure of these transmitters will not result in the operators taking action affecting plant safety because alternative verification of system flow can be indirectly determined through pump load indication via motor ammeter indication, system lineup via valve position check valve open indications from ZS-2118 and ZS-2138, and maintaining reactor vessel level. These indications are displayed in the control room.

4) If coolant injection is not adequate, reactor vessel level will start to drop. This will be indicated in the control room. Upon detection, the operator would take action to provide other means of coolant injection from either the redundant train or some other emergency core cooling system.

36. FLOW TRANSMITTERS FOR LPCI/RHR SYSTEM FLOW INDICATION (BARTON MODEL 368)

a. Action Item Description

LPCI/RHR flow is monitored by flow transmitters FT-1971A and FT-1971B. These transmitters are Barton Model 368 and are located in the RHR corner rooms in panels 1C-129A (FT-1971A) and 1C-129B (FT-1971B). They must be qualified for a total dose of 2.8×10^6 rads (40-year normal plus 30-day LOCA). Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required. Documentation of qualification is not available.

b. Resolution

These instruments will be replaced with flow transmitters during or before the Cycle 8 refueling outage qualified in accordance with 10 CFR 50.49 requirements.

c. Justification for Continued Operation

Continued plant operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) These transmitters perform no automatic safety function; therefore, failure of these instruments will not result in a failure of an automatic safety system to perform its safety function.

3) Failure of these transmitters will not result in the operators taking action affecting plant safety because alternative verification of system flow can be determined through pump load via motor ammeter indication, system lineup using valve position check valve open indications from ZS-1906 and ZS-2002, and maintaining reactor vessel level. These indications are displayed in the control room.

3) If coolant injection is not adequate, the reactor vessel coolant level will start to drop. This will be indicated in the control room. Upon detection, the operator would take action to provide other means of coolant injection from either the redundant train or some other emergency core cooling systems.

37. TEMPERATURE ELEMENTS FOR RHR HEAT EXCHANGER OUTLET TEMPERATURE INDICATION

a. Action Item Description

RHR heat exchanger outlet temperature is currently monitored by thermocouple temperature elements TE-1945C and TE-1945E. The manufacturer is NECI;

the model number is not available. These elements are located in the RHR corner room and are required to be qualified for a total dose of 5.9×10^6 rads (40-year normal plus 30-day LOCA). Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required. Documentation of qualification is not available.

b. Resolution

Qualification by analysis has been determined to not be possible because of unavailable model number information and associated lack of traceability of or otherwise identifiable subcomponent materials information. The existing temperature elements will be replaced with temperature elements qualified to 10 CFR 50.49 requirements during or before the Cycle 8 refueling outage via DCR 1161.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) This equipment performs no automatic safety function; therefore, failure of this equipment will not result in a failure of an automatic safety system to perform its safety function.

3) Temperature detectors (both RTDs and thermocouples) are relatively uncomplicated devices and each manufacturer model type is of similar construction and principle of operation. Because several qualified models exist (e.g., Pyco and Conax), additional reliability exists as to harsh environment post-accident operability. Also, in this case, the equipment qualification concern is limited to the post-accident radiation dose. The subcomponents that would be susceptible to

radiation degradation are the terminal blocks and lead wire insulation. Therefore, failure of the temperature element to provide its temperature dependent signal is unlikely because neither the lead wire nor the terminal block are subject to mechanical stress of a level sufficient to result in gross subcomponent failure.

4) Failure of these instruments (to provide a representative temperature-dependent signal) is unlikely as described above and will not result in the operators taking action affecting plant safety because alternative (see Item 5 below) means of determining instrument failure exist.

5) RHR heat exchanger outlet temperature provides information related to the operation of the RHR system. For this reason, the exact value of RHR heat exchanger outlet temperature is not required. Alternative means of monitoring RHR system heat exchanger performance can be determined from a combination of system lineup via valve position, RHR pump motor amperes indication, and RHR service water (heat exchanger tubeside) temperature indication (TE-1945B and TE-1945E).

38. PRESSURE SWITCHES FOR REACTOR VESSEL SAFETY RELIEF VALVE POSITION INDICATION (PRESSURE CONTROLS MODEL A-17-1N)

a. Action Item Description

The following 24 pressure switches are Pressure Controls Model A-17-1N and monitor the position of the reactor vessel safety relief valves (each of the eight safety relief valves is monitored by three pressure switches that provide a signal indicating an open valve using two-out-of- three logic).

PS-4400A,B,C	PS-4404A,B,C
PS-4401A,B,C	PS-4405A,B,C
PS-4402A,B,C	PS-4406A,B,C
PS-4403A,B,C	PS-4407A,B,C

These pressure switches are qualified for drywell design basis LOCA conditions with the exception of the direct effects of containment spray. Prior to qualification testing, spray deflectors were

installed in the test chamber to deflect the direct effects of the test chamber's spray system. The existence or adequacy of drywell structural devices performing a similar function at the DAEC requires investigation.

b. Resolution

The results of a walkdown that was performed during the Cycle 7 refueling outage has resolved this action item.

Pressure switches PS-4400A,B,C; PS-4401A,B,C; and PS-4403A,B,C through PS-4407A,B,C were found to be effectively shielded from direct spray effects by their sealed overhead junction boxes. Pressure switches PS-4402A,B,C were found to be effectively shielded by a combination of overhead sealed condulets, drywell HVAC ducting, and drywell structural components.

39. POSITION SWITCHES FOR SGTS ISOLATION DAMPER POSITION INDICATION (MICROSWITCH MODEL OPD-AR)

a. Action Item Description

The following six position switches monitor SGTS emergency damper positions within a harsh environment and lack adequate environmental qualification documentation: ZS-5825A,B; ZS-5815A,B; and ZS-7602A,B. These position switches are Microswitch Model OPD-AR and are located in the SGTS room. Position switches ZS-5825A,B and ZS-7602A,B must be qualified for a total integrated radiation dose of 2.1×10^6 rads (based on a distance of approximately 21.5 feet from the only major nearby radiation source). Position switches ZS-5815A,B must be qualified for a total dose of 8.0×10^6 rads (based on a distance of approximately 10.5 feet). Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required. Qualification documentation is not available.

b. Resolution

An evaluation of the radiation and thermal aging effects of the subcomponent materials of position switches ZS-5825A,B and ZS-7602A,B was completed in

September 1983. These position switches were determined to be qualified for a total radiation dose of 3.0×10^6 rads and a thermal aging qualification of 30 years.

Position switches ZS-5815A,B will be replaced with switches qualified to the requirements of 10 CFR 50.49 during or before the Cycle 8 refueling outage.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) These position switches perform no automatic safety function; therefore, failure of any of these position switches will not result in failure of an automatic safety system to perform its safety function.

3) Failure of a damper position switch is unlikely to mislead an operator because realistic harsh environment-induced failure modes will be apparent. Potential failures include either short circuits resulting in both "open" and "closed" lights on or open circuits resulting in both lights off; it is unlikely that a failure can occur resulting in the exact opposite damper position indication.

4) Damper positions in an incorrect or unknown position will not result in an uncontrolled or unacceptable radioactivity release because a bypass piping flowpath around the SGTS filters does not exist.

5) Also, position switches ZS-5815A,B indicate damper position on SGTS trains A and B. Failure of these position switches will not result in the operators taking action affecting plant safety because combined flow through the SGTS could be determined by flow indicators (FIC-5828A,B) on control room panels 1C-24A,B.

40. TERMINAL BLOCKS FOR PRESSURE SWITCHES USED TO MONITOR REACTOR VESSEL SAFETY-RELIEF VALVE POSITION

a. Action Item Description

These terminal blocks are 12-point GE Model EB-25 terminal strips and are located within sealed junction boxes (J1211 through J1216, J1218, and J1219). They are required to be qualified for a total integrated radiation dose of 4.3×10^7 rads (includes 40-year normal dose plus 30-days post-LOCA dose). Each of the eight terminal blocks is connected to three pressure switches (all three of that monitor the pressure downstream of one of the eight reactor safety-relief valves). Qualification documentation for this model terminal block is not available.

b. Resolution

The existing terminal blocks will be replaced with terminal blocks qualified to the requirements of 10 CFR 50.49 during or before the Cycle 8 refueling outage.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) These terminal blocks supply electric power to equipment that provides display information only and are not associated with any automatic safety functions. Therefore, failure of any of these terminal blocks will not result in failure of an automatic safety system to perform its safety function.

2) Required radiation doses were calculated using conservative, nonmechanistic models. Mechanistic release models will result in radiation releases that occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

3) The terminal blocks are static devices and are not subject to any continuous or intermittent stress of an amount likely to cause failure. They were installed in 1979 and are located in sealed enclosures that provide protection from

harsh environment effects other than radiation. The terminal blocks are constructed of a filled phenolic material that is unlikely to fail from the effects of 2.5×10^7 rads radiation dose (5 years of normal operation plus 30-day post-LOCA dose).

4) Instrument failure because of terminal block failure will not result in the operators taking action affecting plant safety because alternative indications exist that would allow a check for confirmation of a stuck-open safety relief valve (i.e., relief valve discharge temperature, reactor vessel level, reactor pressure, suppression pool temperature, and suppression pool water level).

41. ASCo SOLENOID VALVE SV-4639 (MODEL 206-832-2RG) AND OKONITE POWER CABLE (600 V CROSS-LINKED ETHYLENE PROPYLENE)

a. Action Item Description

Solenoid valve SV-4639 is the pilot air supply valve for air-operated control valve CV-4639. Valve CV-4639 is the inboard recirculation system sample line isolation valve and a containment isolation valve. The solenoid valve uses an Okonite 600 V cable as a power supply cable. The solenoid valve and the power supply cable are located at an elevation of 800 feet in the drywell. This location has been determined to be subjected to an ambient temperature of approximately 220F, which is higher than the maximum ambient temperature (150F) for the drywell previously assumed in the thermal aging analysis.

The solenoid valve contains ethylene propylene elastomers. The ethylene propylene elastomers and the ethylene propylene cable insulation have a qualified life of approximately 2 years at 220F. The valve's elastomers and the solenoid valve's associated supply cable are at or near the end of qualified life; therefore, the solenoid valve cannot be considered qualified for post-accident operation.

b. Resolution

The ethylene propylene elastomers and the Okonite supply cable will be replaced, with materials qualified to the extent possible and practical (see Section III.D) for this application, during or before the Cycle 8 refueling outage.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

1) Isolation valve CV-4639 is a normally closed valve; to open the isolation valve, solenoid valve SV-4639 has to be energized. Operation of CV-4639 is not required after an accident. If the elastomers of SV-4639 did fail, air leakage through SV-4639 would be through the exhaust port rather than through the port that operates CV-4639; therefore, CV-4639 would remain closed and maintain its containment isolation function.

2) In the unlikely event that isolation valve CV-4639 failed to shut because of failure of valve SV-4639, containment isolation would be maintained because the other isolation valve (CV-4640) that is outside containment would function.

3) The normal reactor sample point is through the reactor water cleanup (RWCU) system. CV-4639 is the isolation valve for the backup sample system that is only operated when the RWCU system is not available. Because CV-4639 is rarely operated, it is extremely unlikely that a LOCA will occur simultaneous with operation of CV-4639.

VI. TER CATEGORY I.b EQUIPMENT ITEMS (QUALIFICATION PENDING MODIFICATION)

TER Category I.b equipment items have been identified as action items in previous submittals to the NRC. Section II provides a description of each unresolved action item, method of intended resolution, schedule, and justification for continued operation.

The TER comments/concerns for each of the 24 Category I.b equipment item cases were evaluated. Except where otherwise noted (by references in the action item column to an explanatory note), reaffirmation of justification for continued operation is provided by the following cross-index of Category I.b equipment item numbers to Section II action item numbers. (References to explanatory notes at the end of this section are also provided in the left-hand column when necessary to advise of incorrect statements/assumptions appearing in the TER.)

TER Equipment Item	Equipment Description (Plant Identification)	Section II Action Item Number
7.	Limitorque Model SMB-3 (dc motor) with motor brake (MO-2239)	19
9.	Limitorque Model SMB-2 (dc motor) with motor brake (MO-1909)	19
10.	Limitorque Model SMB-2 (ac motor) with motor brake (MO-1908, MO-2238, MO-4627, MO-4628) (see Note VI.A)	19
11.	Limitorque Model SMB-5 (ac motor) with motor brake (MO-1905, MO-2003) (see Note VI.A)	19
12.	Limitorque Model SMB-6 (ac motor) with motor brake (MO-2135, MO-2115) (see Note VI.A)	19 (see Note VI.B)
15.	Limitorque Model SMB-2 (ac motor) with motor brake (MO-2000, MO-2137) (see Note VI.A)	19 (see Note VI.B)
18.	Limitorque Model SMB-2 (ac motor) with motor brake (MO-2117) (see Note VI.A)	19

TER Equipment Item	Equipment Description (Plant Identification)	Section II Action Item Number
19.	Electrodyne Model TN-24-400 (MO-1904, MO-2004)	23
25.	ASCo Model 8320A6 (SV-1963, SV-2033, SV-1966, SV-2037)	24 (see Note VI.D)
29.	ASCo Model HB8302C25RU (SV-4640)	24 (see Note VI.D)
31.	ASCo Model HT831665 (SV-5815A,B; SV-5825A,B; SV-5801A,B; SV-7602A,B)	24 (see Note VI.D)
32.	ASCo Model 831665 (SV-4309)	24 (see Note VI.D)
33.	ASCo Model 831665 (SV-4303)	24 (see Note VI.D)
34.	ASCo Model 831665 (SV-4311, SV-4308, SV-4312, SV-4313, SV-4306, SV-4307)	24 (see Note VI.D)
50.	ITT Barton Model 289 (FIS-2111, FIS-2131,	17 (see Note VI.C)
	PDIS-1971A,B)	18 (see Note VI.C)
51.	SK Model 2096518550 (FM-8408A,B,C,D)	10
63.	Barksdale Model P1H-M85SS-V (PS-8404A,B,C,D)	30 (see Note VI.D)
64.	GE Model 552032HKZZ2 (PDT-1947, PDT-2046)	6
87.	Industrial Engineering Equipment Company Model TFZCP15900 (CVI Drawing A7075900) (1S-1061A,B)	28 (see Note VI.D)
88.	Westinghouse motor Type TEFC (1V-AC-11, 1V-AC-12)	14
89.	Siemens motor Model 2CH6-041-1U (1K-25A,B)	12 (see Note VI.D)
94.	Industrial Engineering Equipment Company Model CT32-23 (TE-5805A,B; TS-5836A,B)	27 (see Notes VI.D and VI.E)

TER Equipment Item	Equipment Description (Plant Identification)	Section II Action Item Number
102.	Rosemount Model 104MA23ABBB (TE-4443A,B,C,D; TE-4444A,B,C,D; TE-4445A,B,C,D; TE-4446A,B,C,D; TE-4477A,B; TE-4478A,B; TE-4479A,B; TE-4480A,B)	29
103.	Penn Model A-19ABB-6 (TS-5808A,B)	25

Section VI Notes:

VI.A These motor operators were incorrectly assumed in the TER to utilize Peerless ac motors. In actuality, they utilize Reliance ac motors.

VI.B TER comments and IELP response to these comments for TER equipment Items 12 and 15 are provided below.

TER Comment (TER Equipment Items 12 and 15)

Due to the relatively nonharsh environment at the installed location and the extensive radiation testing performed on Limitorque motorized valve actuators, qualification can be established by experience with the exception of qualified life.

The licensee has not provided for review (applies to TER Equipment Item 12 only), the documentation or a technical basis to support the claim of a 40-year qualified life estimate (Bechtel Chron 6775).

IELP Response

With the exception of the motor brakes, Limitorque operators utilizing motors with Class B insulation have been evaluated for thermal and radiation aging effects. This evaluation considered known aging properties of subcomponent materials (such as electrical insulation, torque switches, position switches, and lubricants) and identified maintenance and surveillance requirements. The results of this evaluation are summarized in Engineering Analysis of Limitorque dc Class B and ac Class B Motor Valve Operators, dated March 1982 (Bechtel Chron 6775) and Bechtel Aging Evaluation Form L200-00B, dated July 8, 1982 (Bechtel Chron 8109). Franklin Research Center did not previously request these documents for review.

Note: Bechtel Aging Evaluation Form L200-00B has been revised for reasons described in Section III.C. See appropriate Limitorque SCEW sheets for revision number, date, and Bechtel chron number.

VI.C TER comments and associated IELP response for TER equipment Item 50 are presented below.

TER Comments

"Reference 3252" applies to the Barton 288A and 289A; "Reference 3468" is a summary lacking the details needed by an independent reviewer to draw conclusions; licensee has not provided Bechtel Aging Evaluation Form I204-04 (7-8-82), Bechtel Chron 8107 for review and evaluation. It is concluded that the Model 288 or 289 has not been analyzed or tested and, therefore, lacks qualification documentation.

IELP Response

"Reference 3252" (ITT Barton's Report R3-288A-1, dated May 9, 1980, IEEE Standard 344-1975 Seismic and Radiation Qualification Tests on ITT Barton's Differential Pressure Indicating Switches Models 288A and 289A) applies to Model 289 as well as to Model 289A on the basis of similarity. The only difference between Models 289 and 289A is a metal clip added to reduce setpoint drift [see EDS Problem File 0460-067-002 (Bechtel Chron 6863) and QSR-029-A-01 (Bechtel Chron 7719)]. The presence of the clip (which is not susceptible to radiation damage) does not affect the report's applicability to Model 289.

"Reference 3468" is not identified in the TER. This is assumed to be "Reference 5083," BWR Equipment Qualification Summary, ITT Barton Model 289A Differential Pressure Switches, September 23, 1980, QSR-029-A-02. A copy of this document was inadvertently transmitted to the NRC for review in lieu of BWR Equipment Qualification Summary, ITT Barton Model 289, dated October 9, 1980, QSR-029-A-01, because of similarity in report identification numbers and Barton model numbers.

The following critical aging susceptible subcomponent materials have been identified within the Model 289.

PVC lead wire insulation
Phenolic switch
Viton O-ring
Hydrocarbon oil fill fluid
Neoprene gasket

IELP Response (continued)

These subcomponent materials have been evaluated for radiation and aging effects as documented on Bechtel Aging Evaluation Form I204-04, Revision 1, dated January 16, 1983. This aging evaluation establishes a 40-year qualified life and a radiation level qualification of 3.0×10^6 rads. The results were determined by applying the Arrhenius model to the above Model 289 subcomponent materials.

A maintenance/surveillance program is being initiated for the Model 289 to assure performance degradation is minimized from thermal aging of the hydrocarbon oil.

Aging Evaluation Form I204-04 was revised on August 13, 1983, to further clarify surveillance requirements (see Section III.C).

The ITT Barton report, Reference 3252, in conjunction with Aging Evaluation Form I204-04 provides sufficient documentation of ITT Barton Model 289 qualified life and level of radiation qualification.

VI.D This action item has been recently resolved as described in the resolution portion of Section II for this action item; therefore, justification for continued operation for this equipment item is no longer considered necessary.

VI.E Activities related to resolution of this action item have determined that these temperature switches were manufactured by Essex Controls. They were supplied by Industrial Engineering Equipment Company (as Type CT-32-23) to CVI (who supplied the equipment for the DAEC).

VII. TER CATEGORY II.a EQUIPMENT ITEMS (QUALIFICATION NOT ESTABLISHED)

Classification of equipment in this category was the result of FRC's determination that insufficient documentation was provided for review.

The comments/concerns for each of the 56 TER Category II.a equipment items were reviewed as described below and in the following subsections. (Where necessary, a background statement or paragraph is provided to support understanding of the TER comments.)

To simplify this response, Category II.a equipment items have been arranged in generic groups. For example, each of the ten Limitorque equipment items received similar comments regarding unavailable EQ documentation; therefore, the responses have been generically provided.

For 28 of the 56 TER equipment items, documentation presently exists in the DAEC environmental qualification central files which resolves the TER comment/concern; therefore, a response was developed which both identifies the document (by title and a document control number) and summarizes the document's content relative to its method of resolution.

The remaining 28 TER equipment items include DAEC equipment tentatively identified as safety display instruments in previous NRC submittals. Finalization (to the extent now possible) of the list of DAEC accident monitoring instruments requiring environmental qualification is described in Section IV. Using the results of the Section IV approach, 12 of the remaining 28 TER equipment items were determined to be both included in the list of Section IV accident monitoring instruments and to require additional environmental qualification documentation (see Subsections VII.L, VII.M, and VII.N). In each of these cases, a reference to a Section II action item number, description, method of intended resolution and schedule, and justification for continued operation is made. The remaining 16 TER equipment items (see Subsection VII.O) do not satisfy the present criteria for accident monitoring instruments and no longer require environmental qualification.

A. LIMITORQUE MOTOR OPERATORS

TER Comment 1 (TER Equipment Items 1, 2, 5, 6, 8, 14, 16, 121, 122, and 124)

The licensee has not provided documentation from the manufacturer that states the cited test reports are applicable to these equipment items.

IELP Response to TER Comment 1

Equipment Items 1 and 2 address Limitorque motor-operated valve actuators equipped with ac motors using Class H insulation. Equipment Items 14, 16, and 124 address Limitorque motor-operated valve actuators equipped with ac motors using Class B insulation. Equipment Items 5, 6, 8, 121, and 122 address Limitorque motor-operated valve actuators equipped with dc motors using Class B insulation. A review of DAEC environmental qualification records indicates that equipment Item 124 (MO-2701) was in error and should indicate MO-2701 as a dc motor with Class B insulation. The SCEW for MO-2701 has been revised and this equipment item should now be addressed accordingly.

Because Limitorque motor operators are all similar in principle of operation, each Limitorque test report is supportive (to some extent) of motor operator qualification. Limitorque motor operators differ in size, position switch/torque switch materials, and type of motor and insulation class. Limitorque motor operators of all available sizes are generically qualified as described in Limitorque Report B00058. Differences in position switch and torque switch materials (which are located within the motor operator's weatherproof housing) are addressed by an evaluation of radiation and thermal aging effects (which is documented on the Bechtel Aging Evaluation Form referenced on the SCEW sheet). Motors of a given type and insulation class are qualified by the appropriate Limitorque test report as summarized below and in the following subsection.

Test Report	FRC Reference	Motor Type	Insulation Class
B0003	662	ac	B
B0009	1063	dc	H
600376A	1064	ac	H
600456	706	ac	H
B0027	2876	ac	H

1) TER Equipment Items 1 and 2 (ac Class H)

The test reports summarized above show that motors of a similar type and insulation class have been qualified by type test. In addition, for cases where a test report utilizing a motor with Class B insulation is referenced, Class H insulation is considered qualified to the identical environmental conditions for which a Class B insulated motor is qualified because the materials of construction of

Class H insulation are equal to or superior to those used in Class B insulation for radiation and thermal transient conditions. This is supported by a similar statement made by Limitorque Corporation when addressing the like situation for other motors in its letter to Iowa Electric dated April 22, 1982 (Bechtel Chron 7555).

Please note that the TER evaluation section (5f) for these equipment items quoted material in error from our action Item 16 in the January 15, 1982, semi-annual report on environmental qualification (TER Reference 15). This action item is applicable only to dc motors with Class B insulation and should not be referenced for ac motors with Class H insulation.

2. TER Equipment Items 14 and 16 (ac Class B)

Limitorque Corporation, in a letter to Iowa Electric dated April 22, 1982 (Bechtel Chron 7555), has stated that Qualification Report B0003 is applicable to the motor operators included in these equipment items. The applicable SCEWs also reference Bechtel Engineering Analysis of Limitorque Insulation Class B Motor Operators, dated March 26, 1982 (Bechtel Chron 6775). This analysis was performed specifically to provide the documentation establishing the similarity between the motors tested in the test reports (see Section A) and the motors considered qualified by the reports.

Please note that the TER evaluation section (5f) for these equipment items quoted material in error from our Action Item 16 in the January 15, 1982, Semi-Annual Report on Environmental Qualification (TER Reference 15). This action item is applicable only to dc motors with Class B insulation and should not be referenced for ac motors with Class B insulation.

3. TER Equipment Items 5, 6, 8, 121, 122, and 124 (dc Class B)

The applicable SCEWs for these equipment items reference Bechtel Engineering Analysis of Limitorque Insulation Class B Motor Operators, dated March 26, 1982 (Bechtel Chron 6775). This analysis was performed specifically to provide the documentation establishing the similarity between the motors tested in the test reports (see Section A) and the motors considered qualified by the reports.

TER Comment 2 (TER Equipment Items 1, 2, 5, 6, 8, 14, 16, 121, 122, and 124)

The licensee has not provided for review, the documentation or the technical basis to support the claim of a 40-year qualified life estimate.

IELP Response to TER Comment 2

The required documentation for motor operators with Class H insulation motors was documented in Bechtel Aging Evaluation Form L200-00H, dated July 8, 1982 (Bechtel Chron 8109).

The required documentation for motor operators with Class B insulation motors was documented in Bechtel Aging Evaluation Form L200-00B, dated July 8, 1982 (Bechtel Chron 8109).

These two aging evaluation forms document review of thermal aging and radiation test data from the Limitorque test reports and application of the data to DAEC equipment locations. Valve operators with Class B and Class H insulated motors were originally analyzed for maximum (and assumed continuous) normal ambient temperatures of 104 and 150F, respectively. Maintenance requirements to support qualified life conclusions were based on manufacturer recommendations. Four cases of local ambient temperatures above 150F were discovered (two valve operators at 160F maximum, one at 180F, and one at 240F). As a result, the aging evaluation forms were revised to reflect this information and for reasons described in Section III.C.

TER Comment 3 (TER Equipment Item 1)

On Page 3a of this review, the licensee has made reference to "Remark 22" which we have been unable to locate in the documentation submitted.

IELP Response to TER Comment 3

The reference to "Remark 22" was in error in the September 3, 1981, response to NRC Safety Evaluation Report. The remarks column should have been blank for these items; none of the remarks given for Appendix A of that response are applicable. This is confirmed by review of Paragraph IV.A.1.d of the September 3, 1981, response which defines the purpose of the remarks column as follows:

> Reference is made in the remarks column to notes which specifically address the deficiencies identified for each component in the SER. For all components identified as deficient for aging, refer to Section III-3.7 for a discussion of the aging program.

Because these components were identified in Appendix C of the June 3, 1981 SER as deficient only for aging, no remark was required.

B. BARTON MODEL 289 PRESSURE SWITCH

ITT/Barton Model 289, TER Equipment Item 49

TER Comments (NRC Qualification Category)

Reference 3252 applies to the Barton 288A and 289A; Reference 3468 is a summary lacking the details needed by an independent reviewer to draw conclusions; licensee has not provided Bechtel Aging Evaluation Form I204-04 (7-8-82), Bechtel Chron 8107 for review and evaluation. It is concluded that the Models 288 or 289 have not been analyzed or tested and, therefore lack qualification documentation.

IELP Response

However, "Reference 3252" (ITT Barton's Report R3-288A-1, dated May 9, 1980, IEEE Standard 344-1975 Seismic and Radiation Qualification Tests on ITT Barton's Differential Pressure Indicating Switches Models 288A and 289A) applies to Model 289 as well as to Model 289A on the basis of similarity. The only difference between Models 289 and 289A is a metal clip added to reduce setpoint drift [see EDS Problem File 0460-067-002 (Bechtel Chron 6863) and QSR-029-A-01 (Bechtel Chron 7719). The presence of the clip (which is not susceptible to radiation damage) does not affect the report's applicability to Model 289.

"Reference 3468" is not identified in Section 6 of the TER. This is assumed to be Reference 5083, BWR Equipment Qualification Summary, ITT Barton Model 289A Differential Pressure Switches, September 23, 1980, QSR-029-A-02. A copy of this document was inadvertently transmitted to the NRC for review in lieu of BWR Equipment Qualification Summary, ITT Barton Model 289, dated October 9, 1980, QSR-029-A-01, because of similarity in report identification numbers and Barton model numbers.

The following critical aging susceptible subcomponent materials have been identified within the Model 289.

PVC lead wire insulation
Phenolic switch
Viton O-ring
Hydrocarbon oil fill fluid
Neoprene gasket

These subcomponent materials have been evaluated for radiation and aging effects as documented on Bechtel Aging Evaluation Form I204-04, Revision 1, dated January 16, 1983 (Bechtel Chron 10258). This aging evaluation establishes a 40-year qualified life and a radiation level qualification of 3.0×10^6 rads. The results were determined by applying the Arrhenius model to the above Model 289 subcomponent materials.

A maintenance/surveillance program is being initiated for the Model 289 to assure performance degradation is minimized from thermal aging of the hydrocarbon oil.

Bechtel Aging Evaluation Form I204-04 was revised on August 13, 1983, to further clarify surveillance requirements (see Section III.C).

The ITT-Barton report, "Reference 3252," in conjunction with Bechtel Aging Evaluation Form I204-04 provides sufficient documentation of ITT-Barton Model 289 qualified life and level of radiation qualification.

C. BARTON MODEL 763 PRESSURE TRANSMITTER

ITT-Barton Model 763 Transmitter (TER Equipment Item 55)

Background

This equipment was added (DCR 933) for post-accident monitoring of torus water level. It is located in the torus room (north) where qualification for a total integrated radiation dose of 1.3×10^7 rads (normal plus 30-day post-LOCA) is required.

TER Comment

Licensee provided FRC with a copy of ITT-Barton Qualification Test Procedure Document 9999.3154.2. Licensee did not submit ITT Barton Letter 556, dated March 19, 1982; Bechtel Aging Evaluation Form I204-01, dated July 8, 1982; and telecon T. Brendle (Bechtel) to J. Doyen (Barton), dated March 25, 1982 for review and evaluation. FRC concluded that licensee's citations and conclusions are not consistent with ITT-Barton Document 9999.3154.2.

IELP Response

This equipment performs an accident monitoring instrument function. Because of the location in the torus room and post-LOCA accident monitoring function, only post-accident (30-day) radiation dose and aging are required to be considered for environmental qualification.

Previous qualification was based on the documents identified in the TER comment above. These documents (which were not requested for review) utilize test information provided by ITT-Barton relative to radiation and aging qualification and an overly conservative activation energy of 0.6 eV.

Environmental qualification of the Barton 763 model is presently established based on a review of Barton Test Report R3-763-6, dated September 1982 (Vendor Print 11186-212-J-37351-21-1) entitled, ITT-Barton Model 763 Gage Pressure Electronic Transmitter Qualification Test Report. This report demonstrates the adequacy of the Model 763 to perform its specified functions before, during, and after the postulated DAEC service conditions. The tests performed in accordance with IEEE Standard 323-1974 and NUREG 0588 requirements were performed in the following sequence.

1. Accelerated aging (1,830 hours at 125C)

2. Radiation exposure (200 megarads total integrated dose)

3. Seismic simulation (OBE and SSE levels of 9.0 g and 12.5 g, respectively)

4. Design basis event simulation (LOCA and high-energy line break)

Qualified life of the Model 763 transmitter is based on results obtained by applying the Arrhenius model to accelerated aging test parameters. The Arrhenius model requires that the activation energies of the Model 763 transmitter materials be identified and an overall transmitter activation energy be assigned. The activation energies of the materials range from 0.78 eV (metal film resistors) to 1.90 eV (Mica capacitors). Accordingly the limiting activation energy is 0.78 eV. The transmitter with the exception of the transmitter's O-rings (ethylene propylene terpolymer) were subsequently subjected to accelerated aging for 1,830 hours. The O-rings were deemed incapable of withstanding 1,830 hours and were replaced 1,555 hours into the test. Therefore, the O-rings received only 275 hours of accelerated aging. The test's accelerated aging program parameters are summarized below.

Base Parameters

Transmitter test time (except for O-rings), hours	1,830
O-ring test time, hours	275
Test temperature, °C	125 (257F)
Limiting transmitter activation energy (excluding O-ring), eV	0.78
O-ring activation energy, eV	0.95

Applying these parameters to the Arrhenius model, Bechtel Aging Evaluation Form I204-01, Revision 0, documents that the qualified life of the Model 763 transmitter, including O-rings, at the maximum normal service temperature of 104F is at least 40 years.

To address potential dose rate synergistic effects of ethylene propylene resulting from its torus room application, Aging Evaluation Form I204-01 was revised (see Section III.C) on August 10, 1983, to require periodic replacement of the ethylene propylene O-rings every 20 years.

Therefore, the above evaluation in conjunction with the qualification test results of the Model 763 transmitter as documented in Barton Report R3-763-6 establish the transmitters qualification for DAEC application.

D. FENWAL MODEL 350030 TEMPERATURE CONTROL UNIT (TER EQUIPMENT ITEM 66)

Background

Resolution of environmental qualification Action Item 22 (see Section II) states that the electronic controllers of the Fenwal units will be relocated to a mild environment area but the metallic sensors (which are not susceptible to radiation damage) would remain at their present location.

TER Comment

It should be noted that the 30-day operating requirement and 1.6×10^8 rad dose level is significant. The licensee must identify the part number/model number of the metallic sensor and provide evidence to support the position that no radiation-sensitive materials exist.

IELP Response

Each temperature sensor unit consists of a sensor (Fenwal Catalog 35680-4-310 for the 310F units and Catalog 35680-4-255 for the 255F units) and an extension cable assembly (Part 5921-1). An analysis of radiation and thermal aging effects has determined that the limiting subcomponent material of the sensor units is the extension cable insulation (asbestos) which is acceptable for use at radiation doses up to at least 4.8×10^8 rads. This analysis is documented on Bechtel Aging Evaluation Form F081-02, dated October 8, 1982 (Bechtel Chron 10191).

E. GE ELECTRIC HEATER (TER EQUIPMENT ITEM 69)

Background

The GE heater assembly Model 47C518675 was qualified for radiation by analysis (Bechtel Chron 5814). Bechtel Aging Evaluation Form G080-42, dated June 18, 1982 (Bechtel Chron 8105) documents thermal aging qualification.

TER Comment

Heater qualification by analysis lacks technical information such as functional test results, actual test data, anomalies, and deficiencies for an independent reviewer to verify the equipment qualification status. The thermal aging evaluation was not made available for review.

IELP Response

Section 5.1 of the DOR guidelines states that for equipment located in an area requiring post-accident radiation qualification only, radiation qualification may be accomplished by analysis. In Section 5.3, the guidelines provided that in the absence of actual tests, the radiation qualification may be determined by analyzing the effect of radiation environment on the materials used in the equipment. Therefore, radiation qualification may be established by demonstrating that postulated worst-case integrated radiation doses are below the level at which equipment subcomponent materials and subcomponent functionability are affected.

For a piece of equipment as simple in operation as an electric heater, no active, moving function is performed. Therefore, other than continued insulating ability of the extension wire insulation, no other performance characteristics need be considered (the wire insulation is the only organic subcomponent).

The radiation and thermal aging qualification of the heaters was determined in accordance with the above guidelines. In summary, the limiting subcomponent material is polyalkene electric insulation which was found to be qualified for at least 40 years (even when continuously exposed to a maximum normal temperature of 130F) and a total integrated radiation dose of 1×10^8 rads. Requiring functional test results, test anomolies, and test deficiencies for this equipment item application goes beyond the requirements of the DOR guidelines.

F. NAMCO MODEL EA740 POSITION SWITCH (TER EQUIPMENT ITEM 70 AND 71)

Background

Environmental qualification of this equipment is based on NAMCo EA740 Qualification Report, Revision 1, dated February 22, 1979 (Vendor Print E57-1-1, 2-1) and Bechtel Aging Evaluation Form N007-03, dated July 8, 1982 (Bechtel Chron 8111).

TER Comment 1

The qualification test report states that the switch was mounted in a test chamber and attached by means of a threaded pipe (conduit) through which the lead wires were passed. The threaded pipe had been sealed with teflon tape. The test laboratory noted no attempt was made to qualify the connection method. Because licensee has not identified any sealing method or that there is a seal, adequate similarity between equipment and test specimen is not established.

IELP Response to TER Comment 1

To support responses required for NRC Bulletin 79-01B, a walkdown was conducted on March 24 through 28, 1980 (during a scheduled outage), to inspect equipment located in the drywell, steam tunnel, and other locations which are inaccessible during plant operations. The purpose of the walkdown was to determine nameplate information and to identify any equipment conditions that would impact environmental qualification capability. Specific direction provided to the walkdown team members included instruction regarding existence of gaskets, firmness of mounting, and detection of any condition which could result in equipment degrading in a harsh environment. These NAMCo switches are each electrically connected via a tight conduit connection. No abnormalities were detected during the

March 1980 walkdown. Therefore, adequate similarity between DAEC equipment and the NAMCo test specimen is established. Also, DCR 895 installed NAMCo EA-740 position switches; FCR 895-7-0 replaced NAMCo switch original gaskets with silicon rubber gasket. Prior to closure of DCR 895, DAEC maintenance staff verified that conduits were sealed from the limit switch to its junction box with SEMCo PR-855 silicone RTV foam prior to closure of DCR 895 (DCR 895 Package Index Item 41.0-53). The conduit connection itself provides adequate shielding against high-energy line break effects. Completely filling the conduit with silicone RTV foam ensures acceptable level of qualification at this interface.

TER Comment 2

The qualification report states that heat aging was accomplished at 200F for 200 hours; however, the correlation between these conditions and qualified life is not known.

IELP Response to TER Comment 2

Namco has issued a new revision to Qualification Report QTR-111, dated October 1, 1981. This revision continues to be applicable to position switch Model EA-740. The report documents an accelerated aging test of 408 hours at 120C (248F). Namco concludes in the report a qualified life of 7.8 years at 50C (122F) for elastomeric parts using the above accelerated aging data and a conservative activation energy of 0.8 eV. However, Namco also recommends replacement of elastomers at 3-year intervals and the contact block at 20-year intervals.

The current aging evaluation form revision concludes (based on best available industry aging data) a qualified life of 40 years using Arrhenius techniques and a maximum continuous temperature of 150F for each of the nonmetallic subcomponent materials (silicone rubber gasket, ethylene propylene O-ring, and phenolic contact block). The aging evaluation form also concludes that the manufacturer recommended replacement intervals are conservative and reasonable for the DAEC application.

Note that position switch ZS-4639 was replaced with Namco Model EA-180 in 1981. Because the limiting materials in both Models EA-740 and EA-180 are the same, the above conclusions are applicable.

G. LIMITORQUE VALVE OPERATOR POSITION TRANSMITTERS (TER EQUIPMENT ITEMS 76 AND 77)

Background

In References 3 and 4, these position transmitters (ZT-1947 and ZT-2046) were indicated to be qualified based on a knowledge of the equipment's subcomponent materials, function, principle of operation, and verbal input (in the form of a telecon) from Limitorque Corporation. More detailed documentation was being pursued.

TER Comment

Because the telecon is not substantiated by a materials list or detailed drawing submitted by the manufacturer, it cannot be considered documented evidence of qualification.

IELP Response

Subsequent to the final preparation of Reference 4, Limitorque (in response to Iowa Electric Purchase Order 56766, dated March 31, 1982) transmitted a letter (Bechtel Chron 7972) dated June 17, 1982. This letter identified these position transmitters as 25 watt, 1,000 ohm Ohmite Model H potentiometers and provided materials information. Both position transmitters were confirmed to require qualification for harsh environment effects of radiation only; an evaluation of radiation and thermal aging effects determined these position transmitters (which consist of metallic and ceramic materials) to be acceptable for radiation doses up to at least 5×10^9 rads. This analysis is documented on Bechtel Aging Evaluation Form 0026-01, dated September 20, 1982 (Bechtel Chron 10192).

H. NECI MODEL N145C3023 (TER EQUIPMENT ITEMS 104 AND 105)

Background

Post-accident environmental qualification of NECI temperature element Model N145C3023 was based on test data provided in GE's NEDO-24267-1, Supplemental Results of Qualification Data Search for DAEC, dated June 1980 (Bechtel Chron 2572) and the fact that the equipment will perform its safety function (i.e., exceed its setpoint value) prior to exceeding its qualification test value.

TER Comment

The equipment qualification summaries in the GE document lack the technical information needed for an independent reviewer to verify the equipment qualification status. Information such as functional testing results, actual test data, anomalies, deficiencies, and conclusions are lacking; therefore, the GE document cannot be considered evidence of qualification. Also, the establishment of a setpoint temperature and the degree of accuracy for this device does not constitute evidence of qualification for a steam, pressure, and radiation accident environment.

IELP Response

Section 5.1 of the DOR guidelines states that "the choice of qualification method employed for a particular application of equipment is largely a matter of technical judgment based on such factors as: (1) severity of the service conditions; (2) structural and material complexity of the equipment; and (3) degree of certainty required in the qualification procedure."

Section 5.2.5 states that "operational models tested should be representative of the actual application requirements. ... Failure criteria should include instrument accuracy requirements based on the maximum error assumed in the plant safety analysis."

Section 8.0 directs that qualification "records should describe the qualification method in sufficient detail to verify that all of the (DOR) guidelines have been satisfied."

NRC Generic Letter 82-09, dated April 20, 1982, in its attached Clarifications on Environmental Qualification Requirements, Number 8 (1-hour minimum operating time) states that "....test data and analysis may be used to qualify equipment to the required operating time plus an appropriate margin. The 1-hour margin requirement need not be applied. However, subsequent failures should be shown to not be detrimental to plant safety."

These temperature detectors (Type T, copper constantan, dual-element thermocouples) are used for HELB detection in areas of the plant outside of the drywell. A high ambient or differential temperature will cause the appropriate system isolation valve to close, thereby terminating the accident.

These temperature detectors operate exclusively for HELB detection (i.e., they provide no input to any process control system and they are not identified as accident monitoring instruments). Also, the leak detection system, once initiated, will continue to cause system isolation even if the detection variables instrumentation indicates a return to normal. Therefore, failure of these temperature detectors (after performance of their safety function) is neither detrimental to safety nor will it be misleading to the operators.

The GE document (NEDO 24267-1) summarizes the results of functional testing of both elements of a representative (same model) dual-element temperature detector from 40F to approximately 350F. This functional testing was accomplished after exposure to a high temperature (156F) and high relative humidity (90%) environment for 1 hour. During the functional testing, both thermocouple elements performed acceptably with percent error actually decreasing (0.25% to 0.04% and 0.0% to 0.071%) with increased temperature (40.00F to 349.66F). The test summary also provides results of acceptable response time testing.

Postulated HELBs outside the drywell do not exceed 300F temperature, 1.5 psig pressure, and are terminated within 11 seconds by system isolation. Because of the nonexistence of fuel failure, no significant radiation doses result from the HELB. Although this model thermocouple was tested to only 7 inches water column pressure, other similar model thermocouples (such as Pyco's Type T dual-element detector) have been acceptably tested as high as 113 psig. Pressure is not considered a critical environmental qualification parameter for this safety function application and because of the small equipment size and because of the thermocouple's simple principle of operation.

Also, note that radiation dose concerns are limited to that received over a 40-year plant design life (this equipment has no design basis LOCA safety function). Thermal and radiation aging concerns were previously addressed by a conservative surveillance requirement to visually inspect the temperature detector subcomponents every refueling cycle for signs of aging degradation (Reference 3, SCEW 281).

As a result of such an inspection conducted during the 1983 refueling outage, it was determined that this model thermocouple contains no organic materials subject to radiation and thermal aging. The limiting subcomponent

material was found to be a ceramic (similar to steatite) terminal block. Bechtel Aging Evaluation Form N070-02 was revised on August 10, 1983, to incorporate this information (see Section III.C).

The GE document test summary in conjunction with the above analysis demonstrates environmental qualification acceptability of this NECI Model N145C3023 temperature detector. To require a more detailed qualification test report for further evaluation would exceed the requirements of the DOR guidelines as modified by NRC Generic Letter 82-09.

I. PYCO MODEL 02-9039-08-6 (TER EQUIPMENT ITEM 106)

Background

Environmental qualification of this model is based on Pyco Qualification Test Report 770831 (Bechtel Chron 7229), dated August 31, 1977, in conjunction with Bechtel Aging Evaluation Form P427-01 (Bechtel Chron 8115). This Pyco qualification program tested four dual-element RTDs, three single-element thermocouple assemblies, and one dual-element thermocouple assembly. During simulated LOCA testing, Units 1, 3, 4, and 5 exhibited abnormal behavior; Unit 3 ceased operation completely.

TER Comment 1

The model number cited on the licensee's SCEW is not correlated to any test specimen; therefore, there is no information provided in the licensee's submittal which establishes that the items tested is the same model number as is installed in the plant. In addition, the licensee has not stated whether the device is an RTD or thermocouple.

Because of the anomalies which existed during testing, unless traceability to a successful specimen can be made by the licensee, this equipment item is considered deficient regarding acceptable testing.

IELP Response to TER Comment 1

The test report covers qualification tests conducted on Pyco's typical RTD and thermocouple assemblies. The DAEC thermocouple Model 02-9039-08-6 was not specifically tested, but Pyco advises it is similar in all essential respects to the tested thermocouple assemblies. Materials of construction are the same. The singular difference is

that the tested models have spring-loaded and enclosed junctions and Model 02-9039-08-6 has an exposed sealed junction. On the basis of similarity to tested models, Model 02-9039-08-6 was determined to be qualified for DAEC applications.

The testing anomalies indicated in Report 770831 were associated with the following units.

Unit 1: RTD
Unit 3: Thermocouple Type K (chromel-alumel)
Unit 4: Thermocouple Type E (chromel-constantan)
Unit 5: RTD

The test report attributed moisture intrusion as the failure cause. This test anomaly is not considered as affecting thermocouple qualification for the DAEC application because:

The moisture encountered in the simulated LOCA test was created by a chemical spray (3,000 ppm boric acid in a solution with 0.064 molar sodium thiosulfate buffered with sodium hydroxide) whereas only demineralized water moisture is present in the DAEC application. In relative terms of electrical conductivity, the chemical spray is conductive; demineralized water is nonconductive. Accordingly, the intrusion of small amounts of demineralized water moisture would not be expected to affect thermocouple performance. Pyco considers both the test models and Model 02-9039-08-6 moisture-resistant, but because of the sealed junction design, the latter is more moisture-resistant than the former.

Also, the DAEC model is a Type T (copper-constantan) thermocouple (in accordance with vendor Print APED G31-2704). Unit 7 (the only Type T thermocouple tested in the program) performed acceptably during each step or phase of the qualification program. Because only two of the five thermocouples tested exhibited anomalies, no common mode failure concerns were identified; therefore, because of similarity to the Unit 7 test model, the DAEC model thermocouples were determined to be qualified.

TER Comment 2

The test report does not provide a basis for accelerated aging or provide a qualified life. The licensee's SCEW states that the qualified life is 28 years based on Bechtel Aging Evaluation Form P427-01, dated July 8, 1982 (Bechtel

Chron 8115). However, this document has not been provided for review. Because the report does not discuss these details, this item is considered deficient for aging and qualified life assessment based on materials evaluation not being supplied or accomplished by the test report.

IELP Response to TER Comment 2

Pyco's Test Procedure 810713, dated April 5, 1982 (Chron 7393), which is applicable to the DAEC Pyco models, notes that the limiting component is GE SR80 varnish (methyl silicone). The report also gives an activation energy of 0.96 for this varnish. In accordance with Pyco Test Report 770831, dated August 31, 1977 (Chron 7229), test units were aged 168 hours at 121C. Based on the above information, qualified life was conservatively concluded to be 28 years at 104F maximum ambient for these elements using Arrhenius techniques. Pyco is currently in the process of requalifying these temperature elements to NUREG 0588, Category I criteria in accordance with the above test procedure because previous testing was to earlier standards (preliminary indications from Pyco are that the testing phase has been successfully completed and the report will be available the fourth quarter of 1983). Upon receipt of results of the qualification test, the qualified life will be revised if appropriate.

J. GE ELECTRICAL PENETRATION (TER EQUIPMENT ITEM 118)

Background

Previous NRC submittals (References 3 and 4) have identified GE Qualification Test for FO1 Electric Penetration Assembly, dated April 30, 1971 (Bechtel Chron 6898) as a reference for environmental qualification of the DAEC electric penetrations.

TER Comment 1

The test report is for a Type FO1 penetration which is not the same as the installed penetrations, Type NS-02, -03, -04. It should be noted that testing on the NS series penetrations has been conducted by the manufacturer and the licensee should obtain a copy of the applicable report.

IELP Response to TER Comment 1

The Type NS penetrations are Type FO1 penetration. The DAEC electric penetration is of the canister Type FO1 (Models NS02, NS03, and NS04) as established by several documents and correspondence, such as:

1. GE Specification 175A9005, Rev 4, dated July 17, 1969 (Bechtel Chron 6899)

2. GE Letter GHP-7-114, dated December 7, 1977, as referenced by QSR 077-A-01, dated October 12, 1980 (Bechtel Chron 10380)

3. GE Letter G-KE-8-51, dated May 9, 1978 (Bechtel Chron 10476)

The applicability of the FO1 test report was also confirmed by an NRC Region III inspection as described in Report 50-331/78-12, dated June 13, 1978, Docket 50-331, Inspection at Duane Arnold Site, Palo, Iowa. (Inspection was conducted May 10-12 and 17, 1978, by W.D. Schafer and J. Hughes of NRC Region III.)

TER Comment 2

The materials of construction are not described in the report and no evaluation of the susceptibility of the materials to age related degradation is provided in the referenced report. Although the licensee states that an aging analysis was performed (G080-00), the analysis was not provided to permit independent verification.

IELP Response to TER Comment 2

A radiation and thermal aging evaluation was conducted on the DAEC canister type electric penetrations and was summarized in Bechtel Aging Evaluation Form G080-00, dated July 8, 1982 (Bechtel Chron 8105). This aging evaluation form was revised to include four additional electrical penetrations (see Section II, Action Item 20), to provide a more detailed subcomponent analysis, and to further clarify maintenance/surveillance requirements (see Section III.C). Qualification (for aging) of the low-voltage power penetration cable splices remains under investigation as described in Section II, Action Item 20. Justification for continued operation is reaffirmed for this equipment as also described in Section II.

This analysis evaluated the penetration nonmetallic components (epoxy seal, lead wire insulation, and splice/connector insulation materials) for radiation and thermal aging. The limiting material was found to be the nylon-insulated splices (see Section II, Action Item 20 for intended resolution of this potential aging concern). Also, the surveillance requirements identified on the GE electric penetration SCEW sheets (G080-84, -88, and -90) were concluded to be an acceptable means of monitoring potential thermal aging in the epoxy sealing material.

TER Comment 3

No spray testing was performed; it should be noted that the DOR guidelines do not provide for saturated steam as a substitute for chemical spray.

IELP Response to TER Comment 3

DOR guidelines (Section 5.3, Attachment 4, IE Bulletin 79-01B) state that for equipment type-tested for high temperature, pressure, and steam, qualification for chemical spray (e.g., demineralized water) may be demonstrated by analysis. The results of an analysis documenting resolution of this containment spray concern are provided in the resolution description portion of Section II, Action Item 20.

K. ELECTRIC CABLE AND CONNECTORS; RAYCHEM (TER EQUIPMENT ITEM 108): OKONITE (TER EQUIPMENT ITEM 109): ROCKBESTOS (TER EQUIPMENT ITEMS 111 AND 114): ANACONDA ERICKSON (TER EQUIPMENT ITEM 112): AND VICTOREEN CONNECTOR (TER EQUIPMENT ITEM 115)

TER Comment 1 (TER Equipment Item 108)

The licensee should provide the information on the cable insulation thickness (jacket and conductor insulation) and any other characteristics which demonstrate that the installed cable is the same as the cable in the referenced test (F-C4033-1).

TER Comment 2 (TER Equipment Item(s) 108, 109, 112, and 114)

The licensee has not presented sufficient information to establish equivalence between the cable tested and the installed cable as required by DOR guidelines and/or IEEE Standard 383-74.

IELP Response to TER Comments 1 and 2

TER Equipment Item 108

The table below describes the worst-case cables (i.e., minimum insulation and jacket thickness) for Raychem cables purchased for use at the DAEC (Iowa Electric Purchase Order 7884-E-23) and representative cables tested and qualified in Franklin Institute Report F-C4033-3, dated January 1975 (Bechtel Chron 7774). This shows that Raychem cables used at the DAEC are enveloped by this test report.

DAEC Cable Description	Test Sample Description
1. Conductor: #22 AWG, 19-strand tinned copper Dielectric: alkane-imide polymer plus Rayolin R Shield; bare copper wire, #34 AWG having coverage not less than 90% Jacket: black Flamtrol Noisefree treatment: anti-microphonic Outside diameter: 0.242 ± 0.004 in.	Raychem adverse service coaxial cable #22 AWG conductor: first insulation layer, 8 mill wall of alkane-imide polymer; second insulation layer, 49 mil wall of Rayolin R™ radiation cross-linked polyolefin; braided copper shield; Raychem Flamtrol™ jacket, 34 mil nominal wall
2. Conductor: #26 AWG, stranded (7/34), tinned copper First insulation: alkane-imide polymer, nominal diameter 0.027 in. Second insulation: cross-linked cellular polyolefin, nominal diameter 0.285 in. (± 0.007 in.) First shield: tinned copper braid, 90% minimum coverage, nominal diameter 0.306 in. First jacket: flame-retardant, noncorrosive, cross-linked polyolefin, nominal diameter 0.350 in. Second shield: tinned copper braid, 90% minimum coverage, nominal diameter 0.371 in. Second jacket: same as first jacket except nominal diameter 0.437 in. (0.446 in. maximum)	Raychem adverse service triaxial cable, #26 AWG conductor: first insulation layer, 4 mil wall of alkane-imide polymer; second insulation layer, 129 mil wall of Rayfoam F™ radiation cross-linked cellular polyolefin; braided copper shield; first jacket, 22 mil of Raychem Flamtrol; braided copper shield; second jacket 33 mil of Raychem Flamtrol

TER Equipment Item 109

Okonite Letter, J.S. Lasky to J. Hurley, dated June 4, 1980 (Bechtel Chron 1462) states that Qualification Test Report NQRN-1 is applicable to all Okonite cables supplied to the DAEC. Okonite has also indicated in Chron 1462 that although the Okonite insulation used in the test report is very slightly modified from the cables supplied to the DAEC, the modifications are not generic in nature and do not affect the applicability of the test report. In addition, the test cable insulation thickness of 30 mils is less than or equal to the insulation thickness of Okonite cables supplied to the DAEC.

TER Equipment Item 112

The following table lists the worst-case cables (i.e., minimum insulation thickness) of cables listed in Iowa Electric Purchase Orders 51918 and 46332-NG as compared to cables tested in FRC Report F-C4969-1 (Bechtel Chron 7702). The insulation thickness of the tested cables is representative in size to the cables used at the DAEC; therefore, cable qualification is enveloped by Test Report F-C4969-1.

DAEC Cable Description	Test Sample Description
1. Safety-related, Class 1E cable 1/C #14 AWG, seven-strand, copper coated 0.030-inch, FR-EP	600 V ac, FR-EP power and control cable, 1/C, #12 AWG, 7/W, tinned copper conductor, 30-mil insulation thickness
2. Safety-related, Class 1E cable 2/C #16 AWG, seven-strand, coated soft copper 0.025-inch FR-EP insulation, shielded with drain wire, CPE overall, twisted/shielded	600 V ac, instrumentation cable, 2/C, #16 AWG, 7/W, tinned copper conductor, 25-mil flame-resistant, cross-linked EPR jacket insulation (FR-EP), twist, asbestos/mylar tape, tinned copper drain wire, aluminum/mylar tape, 45-mil chlorinated polyethylene jacket (CPE)

TER Equipment Item 114

The following table lists the cables purchased for use at the DAEC (Iowa Electric Purchase Orders 52796 and 46201-NG) and are applicable to Rockbestos Test Reports QR-1804 (Bechtel Chron 7913), QR-1806 (Bechtel Chron 7912), QR-1807 (Bechtel Chron 7911), and FRC Report F-C3798 (Bechtel Chron 10296). The insulation thickness of the cables tested is less than or equal to the insulation thickness of cables used at the DAEC; therefore, the DAEC cable qualification is enveloped by the test reports.

	DAEC Cable Description	Test Sample Description
1.	Safety-related, Class 1E cable, 3/C #16 AWG, seven-strand, coated soft copper, 0.030-inch FR-EP insulation, shielded with drain wire, CPE jacket overall (see Cable Note), twisted/ shielded	From Rockbestos Test Report QR-1804: single-conductor #12 AWG, 600 V, 30-mils of Firewall EPR insulation with 15-mil jacket of Hypalon

Cable Note: The jacket material provides mechanical protection for cable pulling only and is not a critical component for equipment qualification as long as the test sample and actual DAEC cable are both of representative thickness and representative material characteristics (i.e., mechanical durability).

	DAEC Cable Description	Test Sample Description
2.	Safety-related, Class 1E cable, 1/C 250 MCM, 37-coated soft copper, 0.045-inch EP insulation, 0.30-inch Hypalon jacket	Single-conductor #12 AWG, 600 V, 30-mils of strand, Firewall EPR insulation with 15-mil jacket of Hypalon (from Rockbestos Test Report QR-1804)
3.	Safety-related, Class 1E cable, 3/C #16 AWG, Class B stranding, tinned copper, 30-mil flame-retardant cross linked polyolefin insulation color-coded, aluminum, polyester tape shield with tinned copper drain wire, flame-retardant binder in a 45-mil flame-retardant neoprene jacket, rated 90C, 600 V overall, nominal outside diameter of 0.37 inch.	Single-conductor, #16 AWG, 300 V, 20-mils of flame-retardant, chemically cross-linked polyolefin insulation identified as Rockbestos Firewall III; conductor 7/0.0192-inch coated copper (from Rockbestos Test Report QR-1807) and Single-conductor, #12 AWG, 600 V, 30-mils of flame-retardant, irradiation, cross-linked polyolefin insulation identified as Rockbestos Firewall III; conductor 7/0.0305-inch coated copper (from Rockbestos Test Report QR-1806)

	DAEC Cable Description	Test Sample Description
4.	Safety-related, Class 1E cable, 3/C #8 AWG, seven-strand, tinned copper, 45-mil flame-resistant XLPE insulation, 60-mil flame-resistant neoprene jacket	Firewall III: 7/C, #12 TCC, 600 V control cable, 30-mils of flame-retardant XLPE insulation, 45-mils of flame-retardant neoprene jacket (from Franklin Test Report F-C3798)

TER Comment 3 (TER Equipment Item 109)

The licensee has not provided the evaluation of aging degradation for the cable that was referenced on the SCEW sheet. Referenced Okonite Test Report NQRN-1 (Bechtel Chron 1462) has extensive aging data if the licensee can provide traceability to the report.

IELP Response to TER Comment 3

Okonite Test Report NQRN-1 documents accelerated aging results on a graph of time versus temperature for Okonite EP insulation (which is the same type of insulation on the Okonite cables at the DAEC) on Chart 1. Bechtel's Aging Evaluation Form 0004-02 summarizes results of an evaluation of the applicability of Okonite Test Report NQRN-1 to DAEC existing cables. The qualified life of the cables is concluded to be 40 years for maximum continuous temperatures up to 150F. Note: This aging evaluation form was revised on September 9, 1983, for reasons described in Section III.C.

Background for TER Comment 4

Reference 4 identified FRC Test Report F-C4969-1 (Bechtel Chron 7702) as a reference for environmental qualification of this equipment item.

TER Comment 4 (TER Equipment Item 112)

The licensee has not provided the analysis that establishes the equivalence between the thermal preaging conducted in the test program and a 40-year qualified life.

IELP Response to TER Comment 4

FRC Test Report F-C4969-1, Attachment AT-1 documents accelerated aging results on a graph time versus temperature for FR-EP-insulated cable (which is the same type of insulation used on Anaconda-Erickson cable at the

DAEC) on Page 3. Bechtel's Aging Evaluation Form A385-01 summarized results of an evaluation of the applicability of FRC Test Report F-C49659-1 to DAEC existing cables. The qualified life of the cables is concluded to be 40 years for maximum continuous temperatures up to 180F.

Note: This aging evaluation form was revised on September 9, 1983, for reasons described in Section III.C.

TER Comment 5 (TER Equipment Item 111)

The licensee needs to determine from the manufacturer whether the cable installed is suitable for the application. If used in the General Atomics Corporation (GAC) high-range monitor, the cable is not qualified.

IELP Response to TER Comment 5

Rockbestos coaxial cable evaluated on SCEW 338 of Reference 4 was mistakenly identified as having a solid dielectric. This cable was procured by Iowa Electric Purchase Order 57028 and contains a cellular dielectric; therefore, it is qualified by Rockbestos Test Report 2806, Part 2 (Bechtel Chron 7955).

Equipment Item 113 (Rockbestos coaxial cable evaluated on SCEW 335 of Reference 4) was procured in accordance with Iowa Electric Purchase Order 46201 and contains the solid dielectric of concern and is therefore not suitable for high-temperature application (see Note 1 of SCEW 335 of Reference 4). This concern is a known problem resulting in restricted use of this cable at the DAEC. The concern associated with this cable was disseminated via NSAC/INPO Significant Event Report (180), dated February 8, 1982 (Bechtel Chron 6462). Also, GAC high-range radiation monitoring equipment is not used at the DAEC.

TER Comment 6 (TER Equipment Item 115)

The licensee should identify the installed method of connection and justify the integrity of the connection through qualification testing/analysis or document similarity between installed interface and Victoreen Drawing 91007.

IELP Response to TER Comment 6

Note: Drawing 91007 is assumed to be Drawing 910077. The connection procedure (Victoreen Drawing 910077) was used in the installation instructions for these cable assemblies as documented in Iowa Electric Field Change Request 909-1,

Revision 0 (Victoreen Drawing 910077 is Item 5.1 of this FCR package) and therefore, Qualification Report 950.301 [Vendor Print 11186-211-37439-2(6)-1] is applicable to the DAEC-installed equipment.

L. REACTOR SAFETY RELIEF VALVE POSITION INDICATION [TER EQUIPMENT ITEM 60 (PRESSURE CONTROLS MODEL A171N) PLANT ID PS-4400 THROUGH PS-4407, ALL A,B,C]

Background for TER Comment 1

The qualification test [Report 58572 (Bechtel Chron 7314), dated November 12, 1980] conducted by Wyle Laboratories states that the test specimens were installed in a suitable test chamber and that spray deflection plates were installed over the specimens to preclude direct-spray impingement on the specimens.

TER Comment 1

The direct effects of spray and in-leakage have not been considered by the test or licensee.

IELP Response to TER Comment 1

A walkdown was performed to confirm that the DAEC pressure switches are not subjected to the direct effects of containment spray. See Section II, Action Item 38 resolution for more detail.

Background for TER Comment 2

Reference 4 identified the pressure controls Model A171N pressure switch to be qualified for 40 years based on Bechtel Aging Evaluation Form P381-01, dated June 18, 1982 (Chron 8114). This aging evaluation form referenced accelerated aging test data from Qualification Test Report 58572, dated November 12, 1980 (Chron 4413).

TER Comment 2

A deficiency exists with respect to a suitable basis and justification for concluding that 257F for 100 hours establishes a 40-year qualified life. Bechtel Chron 8114 was not submitted for review and evaluation.

IELP Response to TER Comment 2

Bechtel Aging Evaluation Form P381-01 identifies the limiting subcomponent materials to be the kapton-insulated lead wires and the glass-filled phenolic separators. When

using Pressure Controls test report aging test data, an activation energy of 1.57 eV and a maximum continuous normal temperature of 165F, qualified life of at least 40 years is concluded. The Pressure Controls test report does not specifically address aging effects in the phenolic material; therefore, thermal aging information from Plastics, Edition 6, 1983, was used to confirm a qualified life of at least 40 years.

Note: This aging evaluation form was revised on September 9, 1983, for reasons described in Section III.C.

M. CONTAINMENT ISOLATION VALVE POSITION

TER Equipment Item	Manufacturer	Model	Plant Identification
74	NAMCo	SAI31	*ZS-4310
75	NAMCo	SAI31	*ZS-4309
78	NAMCo	SAI131	ZS-4640
82	Microswitch	OPD-AR	ZS-4304, ZS-4305, *ZS-4301
83	Microswitch	OPD-AR	*ZS-4303
84	Microswitch	OPD-AR	ZS-4306, ZS-4307, ZS-4308
85	Microswitch	DTF22RNRH	ZS-3704; ZS-3705; ZS-3728; ZS-3729; ZS-5703A,B; ZS-5704A,B; ZS-5718A,B; ZS-5719A,B
86	Microswitch	DTF22RNRH	ZS-4311, ZS-4312, ZS-4313

The above instruments monitor the position of containment isolation valves. See Section II, Action Item 32 for description of environmental qualification requirements and status, method of intended resolution and schedule, and justification for continued operation.

*Position switches ZS-4310, ZS-4309, ZS-4301, and ZS-4303 have been determined to be located in a mild environment; therefore, environmental qualification is not required.

N. DRYWELL TEMPERATURE

TER Equipment Item	Manufacturer	Model	Plant Identification
95	Leeds & Northrup	89204050005	TE-4328L
96	Leeds & Northrup	892040400321	TE-4386E,F,G,H,J,K,L,M
97	Leeds & Northrup	819710S	TE-4328E,F,G,H,J,K,M

These instruments monitor drywell temperature. See Section II, Action Item 33 for a description of environmental qualification requirements and status, method of intended resolution and schedule, and justification for continued operation.

NOTE: As a result of the evaluation described in Section II, Action Item 33 resolution, only the TER equipment Item 96 temperature elements are required to perform post-accident drywell temperature monitoring.

O. EQUIPMENT NO LONGER IDENTIFIED AS ACCIDENT MONITORING INSTRUMENTATION

TER Equipment Item	Manufacturer	Model	Plant Identification
52	GE	555111BCAA3ABA	FT-3707, FT-3708
53	GE	551032GKZZ2	PT-2306, PT-2207
54	GE	551032EKZZ2	PT-2126, PT-2106
56	Delaval	31924	LS-3701, LS-3721
57	Delaval	XM33353	LE-3701, LE-3721
67	GE	237X731G001	RE-4448A,B,C,D
73	NAMCo	SAI31	ZS-2211, ZS-2212
79	NAMCo	SAI131	ZS-2234, ZS-2235
80	NAMCo	SAI131	ZS-2435, ZS-2436
81	Microswitch	OPD-AR	ZS-7602A,B; ZS-5825A,B
92	NECI	N145C3044	TE-3724
93	NECI	136B3184	TE-4403, TE-4404
99	Leeds & Northrup	819710S	TE-4328A,B,C,D
100	Leeds & Northrup	892040400321	TE-4386A,B,C,D
101	GE	PN133D9679	TE-4400, TE-4401, TE-4402, TE-4405, TE-4406, TE-4407
120	ITT-Barton	368	PDT-4623

The above instruments included in System 36 (Safety Display Instruments) in previous NRC submittals (References 2, 3, and 4) have been determined to not provide primary accident monitoring information. This determination is based on the approach described in Section IV. For these instruments, environmental qualification is not required and justification for continued operation is not provided.

VIII. TER CATEGORY IV EQUIPMENT ITEMS (DOCUMENTATION NOT MADE AVAILABLE)

The TER Category IV classification was based on environmental qualification documents not made available for review by FRC. In each of these cases the documents were not previously requested.

FRC comments on DAEC equipment in Category IV have been reviewed and addressed below by providing a technical summary (responding to the FRC comments) of the qualification documents.

A. AUTOMATIC VALVE COMPANY (AVCo)

Model C5450-5 Solenoid Valves (TER Equipment Item 24)

1. BACKGROUND (TER COMMENT 1)

Environmental Qualification of this equipment is based on BWR Equipment Qualification Summary QSR-052-A-01, dated September 19, 1980 (Bechtel Chron 7500) and its attachment report, Environmental Testing of MSS/RV Air Control Valves 126-62, dated January 15, 1975; ADS Solenoid Valves (AVCo Model C5450-5) Qualification by Analysis by Bechtel Power Corporation, dated March 30, 1982 (Bechtel Chron 6800); and Aging Evaluation Form A613-01, dated July 8, 1982 (Bechtel Chron 8103).

2. TER COMMENT 1

The licensee has referenced a Bechtel Power Corporation analysis which was not made available for review.

3. IELP TECHNICAL SUMMARY RESPONSE TO TER COMMENT 1

The Bechtel Power Corporation analysis was performed in accordance with IEEE Standard 323-1974 and is summarized as follows. The analysis was performed to address the effects of chemical spray.

The AVCo solenoid valve, Model C-5450-5 will be able to withstand the demineralized water spray environmental requirement based on its inherent watertight NEMA 4 specification design, construction, and existing qualification test data. This NEMA 4 application uses Viton seals that have been successfully qualification tested in other types of solenoid valves.

4. BACKGROUND (TER COMMENT 2)

The SER response, dated September 8, 1981, stated that solenoid valves SV-4400, SV-4402, SV-4405, and SV-4406 are qualified for up to 14 years of normal operation plus an accident radiation dose.

The July 15, 1982, semiannual environmental qualification report stated that these solenoid valves are qualified for a 40-year normal operation plus an accident radiation dose based on replacement of Viton seals after 20 years.

5. TER COMMENT 2

The licensee should resolve conflicting statements concerning the qualified life of these valves.

6. IELP TECHNICAL SUMMARY RESPONSE TO TER COMMENT 2

These AVCo solenoid valves have been qualified by test for a total integrated radiation dose of 3.0×10^7 rads. The 14-year qualified life was based on a 30-day accident dose of 2.3×10^7 rads and a 14-year normal operating dose of 7.0×10^6 rads (the 40-year normal operating dose is 2.0×10^7 rads). These radiation values are based on DAEC UFSAR Section 3.11.

An aging review program was completed for the July 15, 1982, semiannual environmental qualification report. A calculational refinement of the 30-day accident dose resulted in a decrease in the radiation dose from 2.3×10^7 to 6.2×10^6 rads. The 40-year qualified life of these valves documented by the Bechtel analysis (Bechtel Chron 6800) and Aging Evaluation Form A613-01 is based on the revised 30-day accident dose of 6.2×10^6 rads and a 40-year normal dose of 2.0×10^7 rads, which gives a total integrated radiation dose of 2.62×10^7 rads (less than the 3.0×10^7 rad qualification value).

Viton seals (in ASCo solenoid valves) have been determined to be susceptible to failure from radiation when exposed to a dose in the range of 2.3×10^7 to 2.0×10^8 rads. Although the AVCo qualification report qualified Viton seals for at least 3.0×10^7, the Viton seals in the DAEC AVCo models will be replaced after 20 years as an additional degree of conservatism.

Replacement of drywell equipment Viton seals after 20 years will limit the total integrated radiation dose (20-year normal plus 30-day accident dose) to less than 2.0×10^7 rads.

Aging Evaluation Form A613-01 was revised on August 10, 1983, to reflect a higher than previously assumed ambient temperature (165F instead of 150F). The higher temperature was determined to not affect the equipment qualified life as described above.

B. ASCO MODEL NP8323A36V SOLENOID VALVES (TER EQUIPMENT ITEMS 47 and 48)

1. BACKGROUND

These solenoid valves are environmentally qualified by ASCo Test Report AQR-67368 Revision 0 dated March 2, 1982 (Bechtel Chron 7411), and AEF A499-00V, dated July 8, 1982 (Bechtel Chron 8102).

2. TER COMMENT

The licensee did not provide ASCo Test Report AQR-67368 or AEF A499-00V for review.

3. IELP TECHNICAL SUMMARY RESPONSE

ASCo Test Report AQR-67368 tested several families of nuclear class (NP series) solenoid valves, including the family containing Model NP8323A36V. Testing of these valves was conducted in accordance with ASCo Qualification Specification AQS-21680, Revision C, which was written to comply with the requirements of the following documents.

a. IEEE Standard 323-1974
b. IEEE Standard 344-1975
c. IEEE Standard 382-1980
d. IEEE Standard 627-1980

These valves were thermally aged at 250F for 18-1/4 days (36-1/2 days for the solenoid coils), wear-aged to 20,100 cycles, radiation-aged to 23 megarads, vibration-aged, and seismically qualified (OBE).

Design basis event (DBE) testing was performed in three phases: seismic DBE simulation, radiation DBE simulation to 182 megarads, and environmental DBE simulation for 30 days at a maximum temperature and pressure of 448F and 68 psig. Chemical sprays consisting of demineralized water and borated water spray in solution with sodium thiosulfate buffered with sodium hydroxide were used. Baseline testing was performed at various stages throughout the test program.

The valve tested for the valve family, including Model NP8323A36V, successfully passed all tests with the exception of the minimum voltage baseline test following the DBE radiation simulation. It was found that the Viton seals used could slightly adhere to the brass seating surfaces at radiation doses exceeding 23 Mrad but less than 200 Mrads.

Bechtel Aging Evaluation Form A499-00V references ASCo Test Report AQR-67368, Revision 0, dated March 2, 1982, as documenting that Viton seals are the limiting component. Aging Evaluation Form A499-00V concludes a conservative qualified life of 20 years for Viton elastomers used in the drywell. Replacement of drywell (and steam tunnel) equipment Viton seals after 20 years will limit the total radiation dose (20 years normal plus 30-day accident dose) to less than 20 Mrads. Therefore, these valves are qualified for 40 years based on replacement of Viton seals after 20 years.

C. STATIC-O-RING MODELS 5NAA3 AND 12NAA5 PRESSURE SWITCHES, TER EQUIPMENT ITEMS 61 AND 62

1. BACKGROUND

These switches are located in an area of the plant which is harsh for radiation only. Reference 4 indicated these switches to be qualified by EDS Analysis (Problem File 0460-067-003), dated March 19, 1982 (Bechtel Chron 6863) and by Bechtel Aging Evaluation Form S382-00, dated July 8, 1982 (Bechtel Chron 8118).

2. TER COMMENT

The licensee has not provided the above references for review; therefore, this equipment item has been placed in Category IV.

3. IELP TECHNICAL SUMMARY RESPONSE

EDS Analysis (Problem File 0460-067-003) summarizes actual test data from a temperature/pressure/humidity test performed on a Model 12N-AA4-TTX10 pressure switch and radiation and thermal aging data for the subcomponent materials used in these switches.

The EDS analysis references Viking Laboratories Test Letter 30203-2, dated November 20, 1973, which documents acceptable operability and repeatability of Static O-Ring Model 12N-AA4-TTX10 under the following simultaneous

environmental conditions: 212F, 7.0 inches water pressure, and 100% relative humidity. This test was repeated for high, low, and midrange pressure settings. Visual inspection of this switch at the completion of the test revealed no major damage.

Bechtel Aging Evaluation Form S382-00 used data from the EDS analysis to establish a qualified life for the switches. Arrhenius calculations were performed for each subcomponent subject to aging degradation. The EDS data showed that the only subcomponent materials with a qualified life of under 40 years at 104F normal temperature is the PVC lead wire with a qualified life of 16 years and the primary diaphragm and O-ring, both Buna-N, with a qualified life of 30 years. The switch materials were found to be qualified for a minimum of 1.0×10^6 rads.

The qualified life of the PVC-insulated lead wires was reevaluated in Revision 1 of Bechtel Aging Evaluation Form S382-00 using PVC aging data from other sources. These other sources included NUREG/CR-2156, Radiation-Thermal Degradation of Polyethylene and Polyvinyl Chloride: Mechanism of Synergism and Dose Rate Effects, dated June 1981, EPRI Report NP-1558, Project 890-1, dated September 1980, and the Industrial Motor User's Handbook of Insulation for Rewinds, by Rejda and Neville, 1977. The dose rate synergistic effect concern associated with PVC (see NUREG/CR-2156) was determined to be not applicable for this DAEC application because of the low 40-year normal integrated radiation dose (i.e., less than 10^4 rads). The qualified life of PVC insulation under conditions where mechanical strength and flexibility (during accident conditions) are not critical to the equipment's safety function, and where the post-accident dose is less than 8×10^6 rads is conservatively determined to be 40 years.

These pressure switches are qualified for 40 years based on replacement of the diaphragms and O-rings after 30 years.

D. <u>BARTON MODEL 288A D/P INDICATING SWITCH (TER EQUIPMENT ITEM 65)</u>

1. BACKGROUND

These switches are located in an area of the plant which is harsh for radiation only. These switches are qualified by Bechtel Aging Evaluation Form I204-06, dated July 8, 1982 (Bechtel Chron 8107).

2. TER COMMENTS

Licensee has referenced Chron 8107 and stated the qualified life of the device is 16 years and that surveillance will monitor the device for degradation at 18-month intervals; however, Chron 8107 has not been provided for NRC review.

3. IELP TECHNICAL SUMMARY RESPONSE

The aging evaluation provided by Bechtel Chron 8107 is based on thermal aging data from EDS analysis [Problem File 0460-067-002 (Bechtel Chron 6863)] and on radiation aging data from ITT/Barton Qualification Report R3-288A-1 (Bechtel Chron 7510).

The EDS analysis lists the service life (based on Arrhenius methodology) of age-susceptible materials used in switch Model 288A. This list shows PVC insulation, with a service life of 16 years to be the limiting material.

The EDS analysis does not provide thermal aging data for the switch's hydrocarbon oil fill fluid. Because thermal aging data are not available, Aging Evaluation Form I204-06 identifies a surveillance requirement to monitor for signs of oil degradation. The performance of the device will be monitored once every refueling cycle by comparing current instrument calibration data to previous calibration; if calibration or setpoint activation exceeds $\pm$ 1-1/2% full scale in the same direction for three consecutive checks or if the hysterisis (deadband tolerance) exceeds 5% of full scale, the switch will be replaced.

Aging Evaluation Form I204-06 was revised to reflect a change in qualified life of PVC insulation from 16 to 40 years (see Section VII.C.3 for details).

ITT/Barton Report R3-288A-1 provides radiation test data on Model 288A switches and for Model 224 hydrocarbon oil fill fluid. The data shows the subject device is qualified for radiation levels up to 3×10^6 rads. This qualification value is in excess of the DAEC's requirement of 2.9×10^5 rads for this application.

E. LOUIS-ALLIS MODEL COG-4B TYPE 19236S-3E371 FAN MOTORS (TER EQUIPMENT ITEM 90)

1. BACKGROUND

These motors are located in an area of the plant which is harsh for radiation only. These motors are qualified by Aging Evaluation Form L280-01, Revision 1, dated August 11, 1983 (Bechtel Chron 12963).

2. TER COMMENT

Aging Evaluation Form L280-01 was not submitted for review.

3. IELP TECHNICAL SUMMARY RESPONSE

Aging Evaluation Form L280-01 is an analysis of radiation and aging degradation for DAEC Fan Motors 1V-EF-15A and 1V-EF-15B (manufactured by Louis Allis).

An analysis of radiation degradation for the fan motor subcomponents shows that the most susceptible subcomponent material (polyester/nylon overcoat) is acceptable for use at exposures up to at least 1.0×10^7 rads.

A thermal aging degradation analysis of these motors based on the similarity to other Class B motors (and using the definition of Class B insulation from IEEE Standard 117-1974) and thermal degradation using Arrhenius methods shows the qualified life for these motors is at least 40 years.

F. GENERAL ELECTRIC MOTORS MODELS 5K6336XC213A and 5K6336XC229A (TER EQUIPMENT ITEMS 91 AND 119)

1. BACKGROUND

These DAEC motors are located in an area of the plant where qualification for radiation (5.9×10^6 rads) is required. Model 5K6336XC229A is qualified by Aging Evaluation Form G080-46, dated June 18, 1982 (Bechtel Chron 8105), Model 5K6336XC213A is qualified by Aging Evaluation Form G080-45, dated June 18, 1982 (Bechtel Chron 8105). Both Aging Evaluation Form G080-45 and Aging Evaluation Form G080-46 are based on GE Qualification Report NSE-76-1281, dated February 8, 1982 (Bechtel Chron 6599).

2. TER COMMENTS

The licensee has not provided Aging Evaluation Form GO80-45, Aging Evaluation Form G080-46 or GE Qualification Report NSE-76-1281 for review.

3. IELP TECHNICAL SUMMARY RESPONSE

Bechtel Aging Evaluation Form G080-46 provides a summary of the acceptability of the aging evaluation results provided in GE Qualification Report NSE-76-1281. GE Report NSE-76-1281 demonstrates qualification of the DAEC motors by similarity analysis to GE test motors, Models 5K6339XC166A and 5K6339XC94A. The DAEC motors are

basically the same construction as the test motors except for size. Size differences between motors will not affect the radiation and thermal aging evaluation results because the insulation system (Class B) used in the motors is the same. The report also:

a. Compared DAEC motors to GE formettes and motors tested to a postulated life, radiation, seismic, steam line break, DBA and post-DBA

b. Reviewed operational experience on GE motors for Class 1E application

c. Performed a failure analysis to show qualified life based on failure data over 15,000 motors

The report shows that the motor materials affected by radiation used can withstand dosages of 4.6×10^7 rads which is in excess of DAEC requirements.

G. GULTON INDUSTRIES MODEL TCA-0646 TEMPERATURE ELEMENT (TER EQUIPMENT ITEM 98)

1. BACKGROUND

These temperature elements are located near the standby gas treatment filters and are harsh for radiation only. Environmental qualification for this equipment is documented by Bechtel Aging Evaluation Form G315-01, dated June 29, 1982 (Bechtel Chron 8106).

2. TER COMMENT

The licensee has not submitted Aging Evaluation Form G315-01 for review.

3. IELP TECHNICAL SUMMARY RESPONSE

Bechtel Aging Evaluation Form G315-01 states that these temperature elements contain no subcomponents which are susceptible to radiation or thermal aging degradation. The only nonmetallic subcomponent material is compressed magnesium oxide which, because of its ceramic properties, is subject neither to thermal aging degradation nor radiation-induced damage.

Bechtel Aging Evaluation Form G315-01 was revised on August 11, 1983, to reflect a higher SGTS filter post-LOCA dose of 3.5×10^8 rads. (The calculation providing the original required dose of 1.6×10^8 rads was found to be

in error and nonconservative for equipment within 1 foot of the filter.) The revised aging evaluation form also concludes a qualified dose of at least 5.0×10^{10} rads based on the ceramic (nonorganic) subcomponents of the temperature elements.

H. KERITE 5KV CABLE MODEL HT WITH NS JACKET (TER EQUIPMENT ITEM 107)

1. BACKGROUND

These cables are environmentally qualified by LOCA Qualification Test of Kerite HTK/HTNS Nonshielded Power Cables, dated February 17, 1981 (Bechtel Chron 7591), and Bechtel Aging Evaluation Form K080-01, dated June 18, 1982 (Bechtel Chron 8108).

2. TER COMMENT

The licensee did not provide the test report for review.

3. IELP TECHNICAL SUMMARY RESPONSE

This is a test report specific to the DAEC which tested a single conductor, #6 AWG, 5 kV nonshielded power cable, 125 mils HTK (N-98) insulation with 80 mils of HTNS (HI-70) jacketing. The test was performed in accordance with IEEE Standard 383-1974 and IEEE Standard 323-1974. This cable was preaged for 100 hours at 150C, irradiated to 200 Mrads, installed in a pressure vessel and subjected to a 100 day steam/chemical spray environment while continuously energized at 2,700 V ac.

This report concludes that Kerite 5 kV, HTK, HTNS jacketed cable can operate while being exposed to harsh environment conditions. Test data can be found in Isomedix Report I-R 975-01, dated October 1975.

IX. TER CATEGORY II.C EQUIPMENT ITEMS (EQUIPMENT SATISFIES ALL REQUIREMENTS EXCEPT QUALIFIED LIFE OR REPLACEMENT SCHEDULE JUSTIFIED)

Classification of equipment in this category was the result of FRC's determination that the equipment satisfies all requirements except qualified life or replacement schedule.

The comments/concern for each of the 16 TER Category II.C equipment items were reviewed as described below and in the following subsections (where necessary, a background statement or paragraph is provided to support understanding of the TER comments).

To simplify this response, Category II.C equipment items have been arranged in generic groups similar to the approach taken in Sections VII and VIII.

A. LIMITORQUE MOTOR OPERATORS

TER Comment 1 (TER Equipment Items 3, 4, 13, 17, and 123)

The licensee has not provided documentation from the manufacturer which states that the cited test reports are applicable to these equipment items.

IELP Response to TER Comment 1

Equipment Items 3 and 4 address Limitorque motor-operated valve (MOV) actuators equipped with dc motors using Class B insulation. Equipment Items 13, 17, and 123 address Limitorque MOV actuators equipped with ac motors using Class B insulation.

See Section VII.A, IELP Response to TER Comment 1, for the remainder of IELP's response to this TER comment.

TER Comment 2 (TER Equipment Items 3, 4, 13, 17, and 123)

The licensee has not provided for review the documentation or the technical basis to support the claim of a 40-year qualified life estimate.

IELP Response to TER Comment 2

See Section VII.A, IELP Response to TER Comment 2, for IELP's response to this TER comment.

B. ASCO MODELS 2068323RV, NP8321A5E, NP8320A173E, AND NP831665E SOLENOID VALVES (TER EQUIPMENT ITEMS 26, 27, 30, 35, 37, 38, AND 39)

Background

Reference 4 identified the above equipment items to be environmentally qualified by ASCo Test Report AQS-21678/TR, Revision A, dated July 1979 (Bechtel Chron 7318).

The more recent ASCo Test Report AQS-67368, Revision 0, dated March 2, 1983 (Bechtel Chron 7411) has been determined to also be applicable to these equipment items.

TER Comment 1

It was concluded that the solenoid enclosure interface degraded to the point where spray solution entered the enclosure, degrading the coil insulation. It is recommended that the licensee provide a suitable seal for the cable entry to the solenoid enclosure.

IELP Response to TER Comment 1

The test report states that the solenoid enclosures were wired through Liquitite Type LT flexible electrical conduit manufactured by Electrical-Flex Company. This conduit is rated for 120F and during the 30-day LOCA test, the plastic liquidtight covering broke down allowing spray solution to enter the solenoid enclosure.

The LOCA simulation test raised the temperature and pressure to 346F/110 psig and maintained conditions above 220F/10 psig for 30 days. However, during and after the test, the solenoid valve was found to operate satisfactorily.

TER Comment 2

The licensee's 20-year replacement interval for ethylene propylene elastomers lacks technical justification. The licensee should present an analysis to support a 20-year qualified life estimate for this equipment.

IELP Response to TER Comment 2

ASCo Qualification Report AQR-67368, Appendix C concluded a 40-year qualified life for ethylene propylene terpolymer (EPDM) when used in the ASCo valve components at 104F continuous ambient temperature. As noted in the report,

ASCo conservatively chose a valve of 0.94 eV as the activation energy and artificially aged the solenoid valves for 18-1/4 days at 250F ambient temperature. Figure 1 of the ASCo qualification report provides a graph of the maximum service periods with respect to ambient temperatures. The 0.94 eV activation energy is conservative because it corresponds to an aging rate associated with retaining 500% elongation retention capability. The activation energy associated with 200% elongation retention is 1.1 eV and would also be conservative.

Bechtel Aging Evaluation Form A499-00E concluded a replacement interval of 20 years for the EPDM elastomer to account for potential (NUREG/CR-2157) dose rate synergistic effects occurring in an oxygen environment. It was concluded that synergistic contributions to aging degradation over a 20-year interval at maximum continuous ambient temperature of 104F would not affect the elastomer's sealing capability.

C. BARTON MODEL 764 LEVEL/PRESSURE TRANSMITTER (TER EQUIPMENT ITEMS 58 AND 59)

Background

The qualified life of 24 years for ITT-Barton Model 764 was established at the time of development of the Reference 3 semiannual report by Bechtel Aging Evaluation Form I204-02 and ITT-Barton Report R3-764-3. Since then, ITT Report R3-764-3 has been superseded with ITT Report R3-764-9 and Bechtel Aging Evaluation Form I204-02 has also been revised.

TER Comment

The licensee has not submitted for review Bechtel Aging Evaluation Form I204-02 (7-8-82), Bechtel Chron 8107.

IELP Response

This equipment performs an accident monitoring instrument function. Because of the location in the torus room and post-LOCA safety display function, only radiation dose and aging are required to be considered for environmental qualification.

Environmental qualification of Barton Model 764 is presently established based on a review of Barton Test Report R3-764-9, dated October 1982 (Vendor Print 11186-212-J-37351-22-1) entitled, ITT-Barton Model 764 Differential Pressure Electronic Transmitter Qualification

Test Report. This report demonstrates the adequacy of the Model 764 to perform its specified functions before, during, and after the postulated DAEC service conditions. The tests performed in accordance with IEEE Standard 323-1974 and NUREG 0588 requirements were performed in the following sequence.

1. Accelerated aging (1,830 hours at 125C)

2. Radiation exposure (200 Mrads total integrated dose)

3. Seismic simulation (OBE and SSE levels of 9.0 g and 12.5, respectively)

4. DBA simulation (LOCA and HELB)

Qualified life of the Model 764 transmitter is based on results obtained by applying the Arrhenius model to accelerated aging test parameters. The Arrhenius model requires that the activation energies of the Model 764 transmitter materials be identified and an overall transmitter activation energy be assigned. The activation energies of the materials range from 0.78 eV. The transmitter with the exception of the transmitter's O-rings (EPDM) were subsequently subjected to accelerated aging for 1,830 hours. The O-rings were deemed incapable of withstanding 1,830 hours and were replaced 1,555 hours into the test. Therefore, the O-rings received only 275 hours of accelerated aging. The test's accelerated aging program parameters are summarized below.

Base Parameters

Transmitter test time (except for O-rings), hours	1,830
O-ring test time, hours	275
Test temperature, °C	125 (257F)
Limiting transmitter activation energy (excluding O-ring), eV	0.78
O-ring activation energy, eV	0.95

Applying these parameters to the Arrhenius model, Bechtel Aging Evaluation Form I204-02 establishes that the qualified life of the Model 764 transmitter, including O-rings, at the maximum normal service temperature of 104F is at least 40 years.

To address potential dose rate synergistic effects of ethylene propylene resulting from their torus room location, Aging Evaluation Form I204-02 was revised to require replacement of the ethylene propylene O-rings every 20 years.

Therefore, the above evaluation in conjunction with the qualification test results of the Model 764 transmitter as documented in Barton Report R3-764-9 established the transmitters qualification for DAEC application.

D. BOSTON INSULATED WIRE MODEL RG6A/U AND 59B/U (TER EQUIPMENT ITEM 110)

Background

Reference 4 identified Boston Insulated Wire (BIW) Qualification Test B913 for cable Types RG-11/U and 59B/U (Bechtel Chron 7640) as a reference for environmental qualification of BIW Type RG-6A/U and 59B/U.

TER Comment

The licensee has not provided an analysis that establishes the equivalency between the accelerated aging described in the test report and the 40-year qualified life identified on the SCEW sheet.

IELP Response

BIW Test Report B913 documents results of thermal aging on equivalent cables on a graph of time versus temperature (Page 12). The criterion for end-of-life of the cables was assumed to be 40% retention of elongation capability which is conservative for electrical supply cable applications. BIW Test Report B913 tested cable Types RG-59B/U and RG-11/U. DAEC cables are RG-59B/U and RG-6A/U. The DAEC cables are equivalent to the cables tested because the three types (RG-59B/U, RG-11/U, and RG-6A/U) all contain BIW fluoropolymer insulation. Aging Evaluation Form B365-01 confirms and concludes a 40-year qualified life for continuous operating temperatures up to 180F.

E. ROCKBESTOS CABLE TYPES RSS-6-104 AND -109 CELLULAR DIELECTRIC (TER EQUIPMENT ITEM 113)

Background

The previous NRC submittal (Reference 4) identified Rockbestos Qualification Test Report 2806 (Bechtel Chron 7955) for cable Types RSS-6-104 and -109 as a reference for environmental qualification.

TER Comment

The licensee has not provided the analysis of aging/ qualified life.

IELP Response

Rockbestos Test Report 2806 documents results of thermal aging for equivalent cables on a graph of time versus temperature (Appendix I, Page 13). The criterion for end-of-life of the cables was assumed to have 60% retention of elongation capability which is conservative for electric supply cable applications. Bechtel Aging Evaluation Form R352-02 confirms the applicability of the Rockbestos Test Report 2806 to DAEC existing cables and concludes a qualified life of 40 years for continuous temperatures up to 180F.