DUANE ARNOLD ENERGY CENTER

SEMIANNUAL REPORT

ON THE

ENVIRONMENTAL QUALIFICATION PROGRAM FOR SAFETY-RELATED

ELECTRICAL EQUIPMENT

IOWA ELECTRIC LIGHT AND POWER COMPANY

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Prepared by
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*Not included in February 4, 1983 report (see Section I)

I. INTRODUCTION

This report provides the third semiannual update on the Duane Arnold Energy Center (DAEC) environmental qualification program. The scope of the report includes an update of efforts to resolve environmental qualification action items, results of efforts to finalize (to the extent possible) the list of DAEC safety display instruments (requiring environmental qualification), and information requested by the Nuclear Regulatory Commission (NRC) safety evaluation report (SER), dated December 20, 1982, on the DAEC environmental qualification program. Sections VI and VII have been added in this report specifically to provide information requested by the SER.

Consistent with the purpose of this report, the master list of Class 1E electrical equipment required to function under postulated accident conditions and the equipment qualification summary sheets [also identified as system component evaluation worksheets (SCEWs)] which were included in previous reports have been omitted. Sections III and V which provide supportive information for the equipment qualification summary sheets have also been omitted.

Section II provides a description of each environmental qualification action item, its intended method and schedule for resolution, and justification for continued operation. Where applicable, the action item resolution schedules were modified to reflect recent changes in schedule for the DAEC refueling outages (fuel cycle 7 changed from Fall 1982 to Spring 1983 and fuel cycle 8 changed from Spring 1984 to Fall 1984). Where equipment replacement is required, resolution completion schedules are indicated relative to a refueling cycle number. For convenience of review, technical changes, additions, or deletions from the corresponding portions of Reference 3 are indicated by revision bars.

Section IV provides a description and status of NRC and Iowa Electric activities affecting finalization of the list of safety display instruments requiring environmental qualification. Iowa Electric's participation in BWR Owners Group activities (related to development of a position on Regulatory Guide 1.97, Revision 2, and development of symptomatic emergency procedure guidelines) has resulted in the DAEC safety display instrument lists (Tables 4-1 and 4-2). For convenience of review, technical changes, additions, or deletions from the corresponding portions of Reference 3 are indicated by revision bars.

The NRC SER dated December 20, 1982, transmitted a technical evaluation report (TER) prepared by Franklin Research Center (FRC). This TER classified DAEC equipment items into one of eight generic qualification categories. The NRC SER requested that justification for continued operation (JCO) be reaffirmed or information submitted for TER equipment items in Categories I.b (qualification pending modification), II.a (qualification not established), and II.b (equipment not qualified) within 30 days of receipt of the SER. The SER also requested that a schedule for proposed corrective actions be submitted within 90 days. Sections VI and VII provide a complete response to both the 30-day and 90-day SER requests as described below (there were no DAEC equipment items identified in Category II.b).

Section VI provides the results of evaluating TER Category I.b equipment item comments. The 24 TER equipment items in this category have been identified as action items in previous submittals. In this section, for each TER equipment item a cross-reference to the Section II action items is made to describe completion of resolution efforts or to reaffirm JCO, and provide the intended method of resolution and schedule.

Section VII provides the results of evaluating TER Category II.a equipment item comments. For the 56 TER equipment items in this category, the information requested by the SER is provided by one of three methods. If the TER comment relates to a piece of equipment which is no longer classified as safety display instrumentation and performs no harsh environment safety function, a reference to Section IV (which provides the safety display instrument determination methodology) is made. If documentation exists in the DAEC environmental qualification central files which resolves the TER comment, the document is cited and briefly described. If documentation does not exist, a reference to a Section II action item is made. Nine new action items (31 through 39) were identified, all of which are safety display instruments and related to efforts of finalization as described in Section IV.

For convenience, Table 1-1 provides a cross-reference from TER equipment item number to the subsection number within this report which addresses the TER comments.

II. ACTION PLAN

The information provided in this section is an item-by-item description of intended IELP 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 justification for continued operation are provided.

1. ASCO SOLENOID VALVES

   Action complete (see Reference 1).

2. MAIN STEAM SAFETY RELIEF SOLENOID VALVES

   Action complete (see Reference 1).

3. TERMINAL BLOCKS

   Action complete (see Reference 1).

4. SOLENOID VALVE SV-4310

   Action complete (see Reference 2).

5. MAIN STEAM ISOLATION VALVE POSITION SWITCHES

   Action complete (see Reference 1).

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 which 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 3)

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

Action complete (see Reference 3)

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

Action complete (see Reference 3)

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 x 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 1984 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 which 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 2).

12. EXHAUST UNITS 1K-25A,B

a. Action Item Description

These motor-operated blowers exhaust MSIV leakage from the leakage control system tank (where it is diluted) to the standby gas treatment system.

The motors are Siemens Model 2CH6-041-1U; however, were supplied by General Electric. They are located in the reactor building above the control rod drive repair room and must be qualified for a total radiation dose of 8.9×10^5 rads. This dose represents a refinement from the previous required value and accounts for the equipment's actual location, distance attenuation from major radiation sources, and design worst-case dilution effects occurring within the MSIV leakage control system.

b. Resolution

Environmental qualification test information from Mississippi Power and Light in conjunction with an engineering analysis completed December 1982 to evaluate radiation and aging effects has determined these blower motors to be qualified for at least 1.0×10^6 rads.

13. MOTOR CONTROL CENTER 1D41

Action complete (see Reference 2).

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. Additional qualification documentation is 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 which 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 3)

16. MOTOR-OPERATED VALVES

Action complete (see Reference 2).

17. FLOW SWITCHES FIS-2111, 2131

a. Action Item Description

These flow switches are Barton Model 289 and are qualified by test for an integrated radiation dose of 3×10^6 rads. These items are located in the RHR rooms and monitor core spray pump discharge flow. They must be qualified for a total dose of 5.9×10^6 rads. (This is the dose at the surface of the most significant source of radioactive fluid contained within the room.)

b. Resolution

Flow switches qualified to NUREG 0588, Category I are not currently available. Because of the relatively minor difference between the required radiation dose and the Model 289 qualification test value, a reevaluation of the required radiation dose at the actual switch location was conducted. Accounting for distance attenuation from the major radiation sources within the room to the switch location, the total required dose is 2.8×10^6 rads (which is less than the qualified value). A radiation and aging evaluation conducted on this equipment model has confirmed its environmental qualification acceptability.

18. PRESSURE SWITCHES PDIS-1971A,B

a. Action Item Description

These pressure switches are Barton Model 289 and are qualified by test for an integrated radiation dose of 3×10^6 rads. These items are located in the RHR rooms and must be qualified for a total dose of 5.9×10^6 rads. (This is the dose at the surface of the most significant source of radioactive fluid contained within the room.).

b. Resolution

Pressure switches qualified to NUREG 0588, Category 1 are not currently available. Because of the relatively minor difference between the required radiation dose and the Model 289

qualification test value, a reevaluation of the required radiation dose at the actual switch location was conducted. Accounting for distance attenuation from the major radiation sources within the room to the switch location, the total required dose is 2.8×10^6 rads (which is less than the qualified value). A radiation and aging evaluation conducted on this equipment model has confirmed its environmental qualification acceptability.

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 which 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 high-energy line break 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
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 housing gasket waterproofing qualified for the post-loss-of-coolant accident (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 DOR Guidelines for their DAEC application.

Environmental qualification of the motor brakes for MO-2000, MO-2115, MO-2135, and MO-2137 will be pursued by engineering analysis of the radiation and thermal aging effects of the motor brake's subcomponent materials. This analysis will be completed by July 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.

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 LPCI injection 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) Valves MO-1902, and MO-2000 are containment isolation valves for the containment atmosphere spray headers in the RHR system. These valves are normally closed and will remain so unless the operator elects to manually initiate drywell or suppression pool spray. If the valves fail 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) Valves MO-2115, MO-2117, MO-2135, and MO-2137 are core spray pump discharge isolation valves. These valves are normally closed and must open to provide core spray flow to the reactor vessel. The environment surrounding these valves does not become harsh until after the valves perform their safety function. The radiation dose to these valve operators is primarily a result of the radioactive process fluid flowing through the valves after they open.

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 933).

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 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 which totally surrounds the penetration and electrical termination area. The enclosure's orientation and adequacy for shielding against water spray and the type of electrical connector and/or splice materials utilized will also be investigated during the Cycle 7 refueling outage. An evaluation of the electrical penetration capability to withstand design basis LOCA effects will be completed by August 1983. Any modifications resulting from the walkdown and/or analysis will be completed by the end of the Cycle 8 refueling outage.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons:

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 reguire 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 3)

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 will be relocated to a mild environment no later than the Cycle 8 refueling outage. Relocation will be via DCR 1121.

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 which 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.

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. These motors and motor brakes will be replaced during or before the Cycle 8 refueling outage with qualified models as follows. The replacement motor will be qualified to NUREG 0588, Category I requirement. The replacement motor brake will be replaced 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 which 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 high-energy line break is postulated. Therefore, the high-energy line break conditions in the torus room have been conservatively applied to the room containing these valves. Due to the remoteness of these valves from the source of the high-energy line break, the actual environmental conditions at the valve's location will be less than specified.

24. ASCO SOLENOID VALVES MODELS HT831665, 831665, 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 which 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 will be replaced during or before the Cycle 8 refueling outage via DCR 1109. The Model 831665 and HT831665 valves will be replaced by ASCO Model NP831665E; the Model 8320A6 will be replaced by ASCO Model NP8320A173E; and the Model HB8302C25RU will be 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.

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 which occur several hours after the postulated accident. Therefore, actual doses will be less than those calculated using such conservative assumptions.

2) The following applies to the identified valves in the SGTS (SV-5815A, B; SV-5825A, B; SV-5801A, B; SV-7602A, B): 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.

3) The following applies to the identified valves in the RHR system (SV-1963, SV-1966, SV-2033, and SV-2037): These solenoid valves control air to their respective control valves to effect isolation of the two independent RHR trains. These valves are normally shut and remain shut to effect system isolation and, therefore, perform no long-term active safety function.

4) The following applies to the identified valves in the containment atmosphere control and primary containment isolation systems (SV-4303, SV-4306, SV-4307, SV-4308, SV-4309, SV-4311, SV-4312, SV-4313, and SV-4640): With the exception of SV-4303 and SV-4309, the function of these valves is limited to containment isolation which occurs in the first few seconds following an accident. These valves need not function following containment isolation. SV-4303 and SV-4309, in addition to the

containment isolation function, may also be required to initiate containment purge within 30 days following an accident. This would occur only in the unlikely event of worst-case hydrogen generation combined with minimum drywell leakage.

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

Because of present unavailability of temperature switches qualified to at least 2.5×10^7 rads, qualification will be pursued primarily by analysis of radiation and thermal aging effects of the existing equipment's subcomponent materials. This analysis is scheduled for completion by August 1, 1983. If environmental qualification by analysis is unsuccessful, the existing temperature switches will be replaced during the Cycle 8 refueling outage with temperature switches qualified to NUREG 0588, Category I 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 which 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 3)

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

a. Action Item Description

Temperature switches TE-5805A,B and TS-5836A,B provide high temperature protection for the SGTS and are located in the SGTS room at elevation 786'. They were manufactured by Essex Controls (Models 351-34912 and 351-253924); however, they were furnished by Industrial Engineering Equipment Company as Model CT32-23. They must be qualified for a total radiation dose of 3.8×10^6 rads. This required radiation dose accounts for distance attenuation from the primary radiation source within the SGTS room. Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required.

b. Resolution

An engineering analysis of radiation and aging effects performed on these switches was completed November 1982. Based on vendor-supplied data, this analysis determined that these switches are qualified for a total radiation dose of 1.2×10^7 rads which exceeds the required dose.

28. INDUSTRIAL ENGINEERING EQUIPMENT COMPANY HEATERS MODEL TFZCP15900

a. Action Item Description

These heaters (1S-1061A,B) were manufactured by Industrial Engineering Equipment Company but supplied by CVI. They function to reduce and/or maintain the relative humidity of the air entering the prefilter in each train of the SGTS and must be qualified for 9.0 x 10^6 rads. This required radiation dose accounts for distance attenuation from the primary radiation source within the SGTS trains. Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required.

b. Resolution

An engineering analysis of radiation and aging effects performed on these heaters to demonstrate environmental qualification was completed September 1982. Based on vendor-supplied data, this analysis determined that these heaters are qualified for a total radiation dose of 1.2 x 10^7 rads which exceeds the required dose.

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 x 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 which 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 which 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 main steam isolation valves in the event of a MSLB. Due to 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

a. Action Item Description

Pressure switches PS-8404A, B, C, and D monitor pressure between MSIVs and provide a low-pressure permit to the MSIV leakage control system. They are Barksdale Model P1H-M85SS-V, but were supplied by General Electric. They are located above the CRD repair room in a location requiring qualification for a total radiation dose of 8.9×10^5 rads. This required radiation dose accounts for distance attenuation and design worst-case dilution effects occurring within the MSIV leakage control system. 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

An engineering analysis of radiation and aging effects completed January 1983, has determined that these pressure switches are qualified for at least 2×10^6 rads.

NOTE: FOR ACTION ITEMS 31 THROUGH 39, VARIABLE TYPES A THROUGH E ARE DEFINED AT THE BOTTOM OF TABLE 4-1.

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

a. Action Item Description

Suppression pool water temperature is a Type A and Type D variable and is currently being monitored by temperature elements TE-4325 and TE-4324. 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 high-energy line breaks 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

Sixteen new temperature elements (TE-4397A through TE-4397H) and TE-4398A through TE-4398H) qualified to NUREG 0588, Category I requirements will be installed to satisfy the requirements of the Mark I containment modification program for monitoring suppression pool water temperature. They will be installed during or before the Cycle 8 refueling outage in accordance with DCR 1137.

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 which 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 high-energy line break 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) High-energy line break postulated in the torus room (i.e., potential break in either the HPCI system steam supply or 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

Primary containment isolation valve position (excluding check valve position) is a Type B variable. 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
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	1.3 E07
ZS-5703B	Microswitch/DTF2-2RN-RH	1.3 E07
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	1.3 E07
ZS-5719B	Microswitch/DTF2-2RN-RH	1.3 E07

The following 20 solenoid valves perform a containment isolation safety function for the primary containment oxygen and hydrogen concentration monitoring system but do not contain position switches.

SV-8101A,B	SV-8106A,B
SV-8102A,B	SV-8107A,B
SV-8103A,B	SV-8108A,B
SV-8104A,B	SV-8109A,B
SV-8105A,B	SV-8110A,B

b. Resolution

For the above identified position switches, environmental qualification will be pursued by analysis of radiation and thermal aging effects of subcomponent materials with completion scheduled for August 1, 1983. Where qualification by analysis is not possible, replacement switches qualified to NUREG 0588, Category I will be installed during or before the Cycle 8 refueling outage.

For the above identified containment isolation solenoid valves, a position switch or some other acceptable method of detecting position will be implemented during or before the Cycle 8 refueling outage. Any new equipment to be installed will meet NUREG 0588, Category I requirements.

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 which 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.

   Lack of position switches on the above solenoid valves will not result in the operators taking action affecting plant safety because solenoid valve "demand" indication combined with process flow indication associated with the hydrogen and oxygen monitoring system can be used to confirm valve position.

4) With the exception of the drywell cooling water system (see Section II.32.c.7.f), 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 switches ZS-3704, ZS-3705, ZS-3728, and ZS-3729 provide position indication for the drywell floor drain sump and the drywell equipment drain sump discharge lines. Additional verification of system isolation is provided by a flow recorder located on control room panel 1C-04.

b) Position switches ZS-4306, ZS-4307, and ZS-4308 provide valve position indication for the torus and drywell fresh air purge system. These valves are designed to fail-closed and require two failures (CV-4306 and CV-4307 or CV-4306 and CV-4308) to cause a loss of primary containment integrity. CV-4307 is normally closed and is likely to remain closed in the event of a LOCA.

c) Position switches ZS-4311, ZS-4312, and ZS-4313 provide valve position indication for the containment atmosphere dilution system. These valves are designed to fail-closed and require two failures (CV-4311 and CV-4312 or CV-4311 and CV-4313) to cause a loss of primary containment integrity. If a loss of indication occurs, additional indication of valve position can be obtained by drywell pressure recorders located in the control room. Valves CV-4312 and CV-4313 are normally closed during operation and are opened only to inert the drywell or torus prior to startup.

d) The following position switches indicate solenoid valve position on the safety-related containment atmosphere dilution system: ZS-4331A, ZS-4331B, ZS-4332A, ZS-4332B, ZS-4333A, ZS-4333B, ZS-4334A, and ZS-4334B. These valves are normally closed and are likely to remain closed in the event of a LOCA (present BWR Owners Group activities have determined that this system will not be needed for hydrogen control post-LOCA and, therefore the containment atmosphere dilution system will remain isolated post-LOCA). Additional indication is provided by PI-4320A and PI-4320B on control room panel 1C-35 which indicate pressure upstream of the solenoid valves. Also, the valves associated with these position switches are backed up by check valves (V-43-82, V-43-84, V-43-86, and V-43-88) which further prevent or reduce the likelihood of uncontrolled radioactivity releases through this system.

e) 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.

f) The following position switches provide control valve position indication for the drywell cooling water system isolation valves: ZS-5703A, ZS-5703B, ZS-5704A, ZS-5704B, ZS-5718A, ZS-5718B, ZS-5719A, and ZS-5719B. The drywell cooling water system is a closed system and does not communicate directly with the drywell atmosphere or penetrate the RCS pressure boundary. Also, the valves associated with position switches ZS-5703A, ZS-5703B, 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.

g) Position switches ZS-4304 and ZS-4305 provide position indication of the suppression pool vacuum breaker line control valves CV-4304 and CV-4305. These valves are normally closed and open only on high differential pressure signals from PDIS-4304 and PDIS-4305. The primary safety display instrumentation purpose of these position switches is to monitor containment integrity. This information is indirectly available via PDIS-4304 and PDIS-4305 alarm signals (these switches provide a signal to either open or shut the associated vacuum breaker line control valve via solenoid valves). Diverse indication of containment integrity is provided by position switches ZS-4329 and ZS-4340 located on the vacuum breakers. It is unlikely (due to differences in location, environmental aging effects, installation-induced mechanical stresses on position switch subcomponents, and post-accident radiation

doses) that all of the position switches in a given line would fail in the same time frame. Because of the low postulated post-LOCA radiation dose (2.9×10^5 rads) in the region of these pressure switches, failure is even more unlikely.

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 a Type D variable and 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 will be conducted to identify the number of temperature elements required to representatively monitor drywell temperature. This number of temperature elements will then be replaced by temperature elements qualified in accordance with NUREG 0588, Category I 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 which 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 pressure is a Type D variable and 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 is classified as Type D because of the information it provides relative to system operation. A more direct indication of MSIV-LCS operation is provided by system flowrate. 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.

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 which 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 a Type D variable and 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 NUREG 0588, Category I 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 which 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 a Type D variable and 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 NUREG 0588, Category I 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 which 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 a Type D variable and is currently monitored by temperature elements TE-1945C and TE-1945E. Information on the manufacturer and 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 of these temperature elements will be pursued by analysis of radiation and thermal aging effects of subcomponent materials with completion scheduled for August 1, 1983. If qualification by analysis is not possible, the existing temperature elements will be replaced with temperature elements qualified to NUREG 0588, Category I requirements 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 which 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 is a Type D variable because it 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

Reactor safety relief valve position is a Type D variable. 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 which 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

A walkdown will be performed during the Cycle 7 refueling outage to investigate the deflection capability of existing drywell structural components. If necessary, metal plates will be installed above the pressure switches during or before the Cycle 8 refueling outage to protect them from direct impingement by containment sprays.

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

2) The primary purpose of these pressure switches is to allow detection of a breach in the reactor pressure boundary caused by a stuck open safety relief valve. In such a case, because containment sprays are not

automatically actuated (i.e., they require manual operator action), operation of the pressure switches in this "nonspray" environment (for which they are qualified) will allow diagnosis of the situation.

3) Instrument failure because of containment spray effects during a LOCA will not result in the operators taking action affecting plant safety for the following reasons.

a) Two-out-of-three logic is required to signal a stuck open relief valve; numerous equipment and structural components exist within the drywell and will act as partial spray deflectors such that it is unlikely that the required two-out-of-three pressure switches will fail due to spray effects.

b) Alternative indications exist which would allow a check for confirmation of a stuck open safety relief valve (these are relief valve discharge temperature, reactor vessel level, reactor pressure, suppression pool water temperature, and suppression pool water level).

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

a. Action Item Description

Emergency ventilation system damper position is a Type D variable. 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. They must be qualified for a maximum radiation dose of 1.6×10^8 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. Qualification documentation is not available.

b. Resolution

Position switches qualified to the requirements of NUREG 0588, Category I will be installed before or during 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 which 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) Additional justification is provided below.

   a) Position switches ZS-5815A,B and ZS-5825A,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.

b) Position switches ZS-7602A,B indicate the position of dampers located between the SGTS and the reactor building ventilation shaft.

These dampers are designed to fail in the open (safe) position. Only one damper is reguired to be open to permit airflow; therefore, both dampers must fail closed and position switches on both dampers must fail to create an unsafe condition. Failure of these switches will not result in the operators taking action affecting plant safety because inadvertent closing of these valves will result in reduced flow on FIC-5828A,B located in the control room. Additionally, these dampers and position switches are located in an environment which is mild for the first hour. In the event of a LOCA, both dampers open automatically and are expected to remain in the open (fail-safe) position.

III. SUPPLEMENTAL INFORMATION

Not included in February 4, 1983 report (see Section I).

IV. SAFETY DISPLAY INSTRUMENTATION

Previous semiannual environmental qualification reports (References 2 and 3) provided a list of safety display instruments and associated environmental qualification summary sheets based on a review of the existing DAEC emergency procedures. References 1 and 2 also indicated that a final list of safety display instruments (requiring environmental qualification) would be completed by January 1, 1983, but that this was dependent on implementation of revised emergency operating procedures during the Fall 1982 refueling outage [subsequently rescheduled to Spring 1983 (see Section I)]. This implementation was in turn dependent on review and approval by the NRC of the symptomatic emergency operating procedure guidelines developed by the BWR Owners Group for NUREG 0737, Item I.C.1.

Efforts to finalize the list of safety display instruments reguiring environmental qualification have been affected by the following activities.

- Issue by the NRC of supplement to NUREG 0737 via Generic Letter 82-33, dated December 17, 1982 (formerly SECY 82-111, Staff Recommendations on Requirements for Emergency Response Capability). This document recommended that licensees develop and implement an integrated program plan for resolution of emergency response capability-related concerns (including emergency operating procedure revisions and safety display instrumentation efforts).

- Issue by the NRC of Generic Letter 82-04, dated April 20, 1982, providing clarifications on environmental qualification requirements of safety-related operator display instrumentation. This letter states that:

  a. All display instrumentation referenced in the emergency procedures need not be identified.

  b. Licensees need only identify and have available qualification documentation on those operator display instruments which are safety-related.

  c. The staff will defer review of the basis for safety classification until other NRC activities [such as preparation of emergency procedures (NUREG 0799), control room design review (NUREG 0700), upgrading of accident monitoring instrument (Regulatory Guide 1.97 and NUREG 0737)] are implemented.

  d. For new or upgraded instrumentation with a required operation date prior to the equipment qualification deadline (of 10 CFR 50.49), qualification must be accomplished by the equipment qualification deadline.

- Submittal by Iowa Electric to the NRC on May 28, 1982, of a 5-year program plan (Reference 5) to integrate engineering, procurement, and installation of planned NRC retrofit requirements and resulting requirements for plant operator retraining with Iowa Electric's own perceived requirements for plant modifications, inspection and maintenance, fuel cycle schedules, and operating plans. This 5-year program plan described Iowa Electric's intentions relative to emergency response facilities.

- Continued participation by Iowa Electric in BWR Owners Group efforts to develop symptomatic emergency operating procedure guidelines for NUREG 0737, Item I.C.1. These guidelines and two subsequent revisions have been submitted to the NRC for review. NRC review and approval was pending in December 1982.

- Finalization of the BWR Owners Group position on Regulatory Guide 1.97, Revision 2. Transmittal to the NRC of this position was approved at the December 1982 BWR Owners Group meeting.

- Issue by the NRC of Generic Letter 82-33, dated December 17, 1982, transmitting Supplement 1 to NUREG 0737. Section 6 of this supplement states that currently installed Regulatory Guide 1.97 instruments are acceptable even if presently not environmentally qualified (if the instrument measures over the range indicated in Regulatory Guide 1.97, Revision 2). However, completion of environmental qualification would eventually be required in accordance with the pending rule (10 CFR 50.49).

- Safety display instrumentation and Regulatory Guide 1.97, Revision 2 continue 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 finalizing (to the extent possible) the list of DAEC safety display instrumentation requiring environmental qualification.

Tables 4-1 and 4-2 contain a list of safety display instruments which has been developed based on this approach and the following considerations.

- Each Type A, B, C, D, and E safety display variable identified in the BWR Owners Group position (and for which Regulatory Guide 1.97, Revision 2, Category 1 or 2 design/qualification criteria are applicable) is listed in Table 4-1 (environmental qualification is applicable only to Category 1 and 2 instruments). For convenience, variables Types A through E are defined at the bottom of Table 4-1.

- DAEC instrument transmitters are identified in Table 4-1 by plant identification number for each safety display variable. Instrument loop components (other than transmitters) such as power supplies, signal conditioners, and control room indicators are not listed in the table because they are in mild environment areas and outside the scope of this report. Containment isolation valve position switches (because of their large number) are listed separately in Table 4-2 along with piping penetration number.

- The equipment in Tables 4-1 and 4-2 are assigned a harsh or mild post-accident environmental status based on plant location and post-accident operability requirements (for harsh environment instruments, manufacturer and model number are also provided).

- The environmental qualification of instruments in Tables 4-1 and 4-2 was evaluated by reviewing previous results summarized on system component evaluation worksheets (SCEW) transmitted with Reference 3 and in light of any applicable comments in the TER. If a deficiency exists, a Section II action item is referenced; otherwise, a Reference 3 SCEW number is referenced which documents the determination of acceptable environmental qualification documentation.

- For the containment isolation valve position variable:

  - Check valves and instrument lines are excluded in accordance with Regulatory Guide 1.97, Revision 2 and proposed regulatory guide (Task IE-126-5), Instrument Sensing Lines

  - Explosion-actuated containment isolation valves (within the transversing-in-core probe system and the standby liquid control system) are excluded; these valves are actuated to their safe position when required. Installation of position switches on these valves is not practical.

  - Isolation valves associated with the control rod drive scram discharge volume piping are not included because the containment isolation boundary is considered to be the control rod drive pistons.

  - Environmental qualification of position switches within Limitorque motor operators are addressed by qualification of the associated operator.

  - Environmental qualification of position switches within the Electrodyne operators (MO-1904, MO-2004) are being addressed with Section II, Action Item 23.

As progress occurs in other safety display instrumentation/ Regulatory Guide 1.97 areas, updates/modifications to the list of safety display instrumentation may result.

V. DESCRIPTION OF REPORT DATA SHEETS AND EVALUATION SHEETS

Not included in February 4, 1983 report (see Section I).

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

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 are also provided in the left-hand column when necessary to advise of incorrect statements/assumptions appearing in the TER.)

TER Equipment Item Number		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 Number		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
29.	ASCo Model HB8302C25RU (SV-4640)	24
31.	ASCo Model HT831665 (SV-5815A,B; SV-5825A,B; SV-5801A,B; SV-7602A,B)	24
32.	ASCo Model 831665 (SV-4309)	24
33.	ASCo Model 831665 (SV-4303)	24
34.	ASCo Model 831665 (SV-4311, SV-4308, SV-4312, SV-4313, SV-4306, SV-4307)	24
50.	ITT Barton Model 289 (FIS-2111, FIS-2131, PDIS-1971A,B)	17 (see Note VI.C) 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)
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 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.

TER Comment (TER Equipment Item 12)

The licensee has not provided for review of 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.

VI.C 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

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.

The ITT Barton report, Reference 3252, in conjunction with 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 in the following subsections, 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 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 possible) of the list of DAEC safety display 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 safety display 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 safety display instruments and no longer require environmental qualification.

A. LIMITORQUE MOTOR OPERATORS

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

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

IELP Response

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 (SCEW 206, MO-2701) was in error and should indicate MO-2701 as a dc motor with Class B insulation. This SCEW is currently being revised and this equipment item should now be addressed accordingly. The SCEWs for the above equipment items reference Limitorque test reports utilizing test specimen motors as follows:

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 referenced in Section A 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)

By inspection, it can be seen that motors of a similar type have been used in qualification tests (see Section A). In addition, Limitorque Corporation, in a letter to Iowa Electric dated April 22, 1982 (Bechtel Chron 7555) has specifically 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 (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

The required documentation for equipment Items 1 and 2 is provided in Bechtel Aging Evaluation Form L200-00H, dated July 8, 1982 (Bechtel Chron 8109).

The required documentation for equipment Items 5, 6, 8, 14, 16, 121, 122, and 124 is provided in Bechtel Aging Evaluation Form L200-00B, dated July 8, 1982 (Bechtel Chron 8109).

TER Comment (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

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 ensure no performance degradation from thermal aging of the hydrocarbon oil.

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 a safety display instrument function. Because of the location in the torus room and post-LOCA safety display 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 1,550 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 establishes 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.

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). A Bechtel Thermal Aging Evaluation G080-42, dated June 18, 1982 (Bechtel Chron 8105) was performed on the heaters to demonstrate 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

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 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

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

The equipment's original gaskets have been replaced with gaskets of longer lasting silicone rubber material (see Reference 1, Action Item 5); an evaluation of radiation and thermal aging effects was accomplished to determine the limiting subcomponent material. This evaluation is

documented in Bechtel Aging Evaluation Form N007-003, dated July 8, 1982 (Bechtel Chron 8111). Previously, the original gaskets were limiting with a qualified life of 1 year (for the worst-case normal temperature location; that is, the drywell at 150F). The current evaluation indicates the silicone rubber gaskets to be limiting with a qualified life of 9 years (using an activation energy of 1.14 eV for silicone rubber).

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

Background

In References 2 and 3, 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 3, 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 high-energy line break 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 high-energy line break detection (i.e., they provide no input to any process control system and they are not identified as safety display 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 high-energy line breaks 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 high-energy line break. 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 are presently being addressed by a very conservative surveillance requirement to visually inspect the temperature detector subcomponents every

refueling cycle for signs of aging degradation (Reference 3, SCEW 281). This inspection will be accomplished until or unless a qualified life is otherwise established. GE is presently pursuing an aging analysis of this equipment in response to Iowa Electric Purchase Order 61747.

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 Comments

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.

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.

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

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. The qualification credentials of Model 02-9039-08-6 will be reaffirmed upon completion (targeted for March 1983) of current qualification testing to the requirements of NUREG 0588, Category I.

J. GE ELECTRICAL PENETRATION (TER EQUIPMENT ITEM 118)

Background

Previous NRC submittals (References 2 and 3) 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

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

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

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.

Also, 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

An initial aging evaluation was conducted on the DAEC canister type electric penetrations as summarized in Bechtel Aging Evaluation Form G080-00, dated July 8, 1982 (Bechtel Chron 8105). Environmental qualification of these electrical penetration remains under investigation with respect to demineralized water spray and limited cable thermal aging of the low-voltage power penetration cable splices as described in Section II, Action Item 20. Justification for continued operation is reaffirmed for this equipment as also described in Section II.

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 (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 (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

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 polymar 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)
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 (TER Equipment Item(s) 109 and 112)

The licensee has not provided the evaluation of aging degradation for the cable which was referenced on the SCEW sheet.

IELP Response

Equipment Item 112 (Anaconda-Erickson 600 V cable) is evaluated for aging degradation in FRC Report F-C4969-1 prepared for Anaconda Company July 1979 (Bechtel Chron 7702) and Attachment AT-1 to FRC Report F-C4969-1, Revision 1, dated June 1981 (Bechtel Chron 7703).

Using a worst-case maximum (assumed continuous) temperature of 150F and the Arrhenius aging data and figures provided in Attachment AT-1 to Report F-C4969-1, a qualified life of more than 40 years results.

TER Comment (TER Equipment Item 109)

Referenced Report NQRN-1 has extensive aging data if the licensee can provide traceability to the report.

IELP Response

TER Equipment Item 109 (Okonite 600 V cable) is traceable to Report NQRN-1 by Purchase Order 7884-E-019; therefore, the aging degradation evaluation contained in the report is applicable. Using a worst-case maximum (assumed continuous) temperature of 150F and Chart 1 of Test Report NQRN-1 (Bechtel Chron 1462), the qualified life is more than 40 years. Chart 1 is based on Arrhenius methodology.

TER Comment (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

Rockbestos coaxial cable evaluated on SCEW 338 of Reference 3 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 3) 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 3). 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 (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

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

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

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

IELP Response

A walkdown will be required to confirm that the DAEC pressure switches are not subjected to the direct effects of containment spray. Until this information is confirmed, Action Item 38 has been assigned. See Section II for a description of this action item, method of intended resolution and schedule, and justification for continued operation.

TER Comment

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

The commitment made on the response to the original environmental qualification SER for establishing an aging review program by June 30, 1981, was met and the results were submitted with Reference 3. The associated aging evaluation forms can be summarized as follows: the limiting subcomponent material is the Kapton lead wire insulation. Assuming a maximum continuous normal temperature of 150F and using the aging test data from Report 58577 (including an activation energy of 1.58 eV) results in a qualified life greater than 40 years.

Bechtel Aging Evaluation Form P381-01 (Bechtel Chron 8114) concluded a qualified life of 40 years based on a review of subcomponent materials, activation energies, aging test data for the limiting material, a normal maximum temperature of 150F, and Arrhenius techniques.

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 and are included in the list of Section IV safety display instruments. 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 reclassified as being located in a mild environment (see Table 4-2); 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 which is included in the list of Section IV safety display instrument parameters. 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.

O. EQUIPMENT NO LONGER IDENTIFIED AS SAFETY DISPLAY 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 1, 2, and 3) have been determined to not provide primary safety display 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. LIST OF REFERENCES

1. Iowa Electric Letter LDR-81-257, L.D. Root to H. Denton, dated September 3, 1981, transmitting the DAEC Response to the NRC Safety Evaluation Report on Environmental Qualification of Safety-Related Electrical Equipment

2. Iowa Electric Letter LDR-82-013, L.D. Root to H. Denton, dated January 29, 1982, transmitting the DAEC Semiannual Report on the Environmental Qualification Program for Safety-Related Electrical Equipment, Iowa Electric Light and Power Company, Docket 50-331, January 15, 1982

3. Iowa Electric Letter LDR-82-191, L.D. Root to H. Denton, dated July 15, 1982, transmitting the DAEC Semiannual Report on the Environmental Qualification Program for Safety-Related Electrical Equipment, Iowa Electric Light and Power Company, Docket 50-331, July 15, 1982

4. Iowa Electric Letter LDR-80-310, L.D. Root to J. Keppler, dated October 31, 1980, transmitting the DAEC Response to NRC IE Bulletin 79-01B Environmental Qualification of Class 1E Equipment

5. Iowa Electric Letter LDR-82-0140, L.D. Root to H. Denton, dated May 28, 1982, transmitting Iowa Electric's Integrated Program for Modification of the Duane Arnold Energy Center

TABLE 1-1

TER EQUIPMENT ITEM CROSS REFERENCE

TER Equipment Item	DAEC Plant Identification	TER Qualification Category*	Third Semiannual EQ Report Subsection	Section II Action Item
001	MO-8401A-D, MO-8402A-D, MO-8403A-D	II.a	VII.A	
002	MO-1900, MO-2400, MO-2700, MO-4423	II.a	VII.A	
005	MO-2321	II.a	VII.A	
006	MO-2312	II.a	VII.A	
007	MO-2239	I.b	VI	19
008	MO-1937	II.a	VII.A	
009	MO-1909	I.b	VI	19
010	MO-1908, MO-2238, MO-4627, MO-4628	I.b	VI	19
011	MO-1905, MO-2003	I.b	VI	19
012	MO-2135, MO-2115	I.b	VI, VI.B	19
014	MO-2046	II.a	VII.A	
015	MO-2000	I.b	VI, VI.B	19
016	MO-1934, MO-2007, MO-1970, MO-2009, MO-2104, MO-2124, MO-2290A,B; MO-2038, MO-1935, MO-2069, MO-1989, MO-2005, MO-1932, MO-2010, MO-2112, MO-2132, MO-1933, MO-1936, MO-2146, MO-2147, MO-1943B, MO-2006	II.a	VII.A	
018	MO-2117	I.b	VI	
019	MO-1904, MO-2004	I.b	VI	23
025	SV-1963, SV-2033	I.b	VI	24
029	SV-4640	I.b	VI	24
031	SV-5815A,B; SV-5825A,B; SV-5801A,B; SV-7602A,B	I.b	VI	24
032	SV-4309	I.b	VI	24
033	SV-4303	I.b	VI	24
034	SV-4311, SV-4308, SV-4312, SV-4313	I.b	VI	24

TER Equipment Item	DAEC Plant Identification	TER Qualification Category*	Third Semiannual EQ Report Subsection	Section II Action Item
049	PS-4348	II.a	VII.B	
050	PDIS-1971A,B	I.b	VI	18
051	FM-8408A-D	I.b	VI	10
052	FT-3707, FT-3708	II.a	VII.O	
053	PT-2207, PT-2306	II.a	VII.O	
054	PT-2126, PT-2106	II.a	VII.O	
055	LT-4396A,B	II.a	VII.C	
056	LS-3701, LS-3721	II.a	VII.O	
057	LE-3701, LE-3721	II.a	VII.O	
060	PS-4400A,B,C through PS-4407A, B,C	II.a	VII.L	38
063	PS-8404A-D	I.b	VI	30
064	PDT-1947, PDT-2046	I.b	VI	6
066	CU-5835A1,A2,B1, B2; CV-5837A1,A2, B1,B2	II.a	VII.D	22
067	RE-4448A-D	II.a	VII.O	
069	1S-122A-D	II.a	VII.E	11
070	ZS-4413, ZS-4416, ZS-4419, ZS-4421	II.a	VII.F	
071	ZS-4412, ZS-4415, ZS-4418, ZS-4420, ZS-1906A, ZS-2002B, ZS-4639, ZS-2002A, ZS-1906B	II.a	VII.F	
073	ZS-2211, ZS-2212	II.a	VII.O	
074	ZS-4310	II.a	VII.M	
075	ZS-4309	II.a	VII.M	
076	ZT-1947	II.a	VII.G	
077	ZT-2046	II.a	VII.G	
078	ZS-4640	II.a	VII.M	32
079	ZS-2234, ZS-2235	II.a	VII.O	
080	ZS-2435, ZS-2436	II.a	VII.O	
081	ZS-7602A,B; ZS-5825A,B	II.a	VII.O	39
082	ZS-4301, ZS-4304, ZS-4305	II.a	VII.M	32
083	ZS-4303	II.a	VII.M	
084	ZS-4306, ZS-4307, ZS-4308	II.a	VII.M	32
085	ZS-3704, ZS-3705, ZS-3729, ZS-5703A,B; ZS-5704A,B; ZS-5718A, ZS-5718B, ZS-5719A,B; ZS-3728	II.a	VII.M	32

TER Equipment Item	DAEC Plant Identification	TER Qualification Category*	Third Semiannual EQ Report Subsection	Section II Action Item
086	ZS-4311, ZS-4312, ZS-4313	II.a	VII.M	32
087	1S-1061A,B	I.b	VI	28
088	1V-AC-11, 1V-AC-12	I.b	VI	14
089	1K-25A,B	I.b	VI	12
092	TE-3724	II.a	VII.O	
093	TE-4403, TE-4404	II.a	VII.O	
094	TS-5836A,B; TS-5805A,B	I.b	VI	27
095	TE-4328L	II.a	VII.N	
096	TE-4386E-H, J-M	II.a	VII.N	33
097	TE-4328E-H, J,K,M	II.a	VII.N	33
099	TE-4328A-D	II.a	VII.O	
100	TE-4386A-D	II.a	VII.O	
101	TE-4400, TE-4401, TE-4402, TE-4405, TE-4406, TE-4407	II.a	VII.O	
102	TE-4443A-D; TE-4444A-D; TE-4445A-D; TE-4446A-D; TE-4477A,B; TE-4478A,B; TE-4479A,B; TE-4480A,B	I.h	VI	29
103	TS-5808A,B	I.b	VI	25
104	TE-2522A-D; TE-2532A-D; TE-2526A-D	II.a	VII.H	
105	TE-2262A,B; TE-2263A,B; TE-2264A,B; TE-2453, TE-4447, TE-4451A,B; TE-2265	II.a	VII.H	
106	TE-2742A-F; TE-2743A-F; TE-2744A-F	II.a	VII.I	
108	Cable-coax-ray	II.a	VII.K	
109	Cable-instr-Oko; cable-power-Oko; cable-control-Oko	II.a	VII.K	
111	Cable-coax2-ROC	II.a	VII.K	
112	Cable-power-A/E; cable-control-A/E; cable-instr-A/E	II.a	VII.K	

TER Equipment Item	DAEC Plant Identification	TER Qualification Category*	Third Semiannual EQ Report Subsection	Section II Action Item
114	Cable-power-ROC; cable-control-ROC; cable-instr-ROC	II.a	VII.K	
115	Instrument cable	II.a	VII.K	
118	JX-103; 105A-D; 101A,B; 104A-D	II.a	VII.J	20
120	PDT-4623	II.a	VII.O	
121	MO-2516	II.a	VII.A	
122	MO-2401, MO-2512	II.a	VII.A	
124	MO-2701	II.a	VII.A	

*Key

I.b	Qualification pending modificaton
II.a	Qualificatin not established

TABLE 4-1

SAFETY DISPLAY INSTRUMENTS

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	[2] Manufacturer/Model
Drywell hydrogen concentration	1	AR-4381A AR-4382A	RB/1C-218A RB/1C-218B	Mild Mild	Type A, C variable; implemented in accordance with NUREG 0737 (II.F.1-6)	
Drywell oxygen concentration	1	AR-4381B AR-4382B	RB/1C-218A RB-1C-218B	Mild Mild	Type A, C variable	
Reactor vessel level	1	LITS-4539 LITS-4540 LT-4541 LT-4559 LT-4560 LT-4561 LITS-4565 LITS-4566	RB/1C-56 RB/1C-55 RB/1C-56 RB/1C-56 RB/1C-55 RB/1C-56 RB/1C-122 RB/1C-121	Mild Mild Mild Mild Mild Mild Mild Mild	Type A, B variable	
Suppression pool water temperature	1	TE-4397A through H TE-4398A through H	Torus Torus	Harsh Harsh	Type A, D variable; being implemented via DCR 1137 for Mark I containment modification program (see Section II, Action Item 31)	Not available Not available
Suppression pool water level	1	LT-4396A LT-4396B LT-4397A LT-4397B	Torus room Torus room Torus room Torus room	Harsh Harsh Harsh Harsh	Type A, C, D variable; implemented in accordance with NUREG 0737 (II.F.1-5); qualified in accordance with Section V.II.C and Reference 3, App B, SCEWs 125 and 127)	ITT-Barton/763 ITT-Barton/764 ITT-Barton/764 ITT-Barton/764
Drywell Pressure	1	PT-4398A PT-4398B PT-4399A PT-4399B	RB RB RB RB	Harsh Harsh Mild Mild	Type A, B, C, D variable; implemented in accordance with NUREG 0737 (II.F.1-4); qualified in accordance with Reference 3, App B, SCEW 127	ITT-Barton/764 ITT-Barton/764

Table 4-1 (continued)

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	Manufacturer/Model[2]
RCS Pressure	1	PT-4599A	RB/1C-55	Mild	Type A, B, C variable;	
		PT-4599B	RB/1C-55	Mild		
Containment isolation valve position indication	1	Various			Type B variable; see Table 2	
Neutron flux	2	RE-4572A	RV	Mild	Type B variable; not required to be qualified for LOCA or HELB effects according to IE Bulletin 79-01B, Supplement 2	
		RE-4572B	RV	Mild		
		RE-4572C	RV	Mild		
		RE-4572D	RV	Mild		
		RE-4572E	RV	Mild		
		RE-4572F	RV	Mild		
		RE-4571A	RV	Mild		
		RE-4571B	RV	Mild		
		RE-4571C	RV	Mild		
		RE-4571D	RV	Mild		
		RE-4571E	RV	Mild		
		RE-4571F	RV	Mild		
		RE-4571G	RV	Mild		
		RE-4571H	RV	Mild		
		RE-4571I	RV	Mild		
Effluent noble gas concentration	2	RE-4176	RB/1C-334A	Mild	Type C, E variable; implemented in accordance with NUREG 0737 (II.F.1-1)	
		RE-4175	RB/1C-333A	Mild		
		RM-4176	RB/1C-334B	Mild		
		RM-4175	RB/1C-333B	Mild		
Drywell atmosphere temperature	2	TE-4386E through H	Drywell	Harsh	Type D variable; see Section II, Action Item 33	L&N/8920-404-00-3-21
		TE-4386J through M				L&N/8197-10-S
		TE-4328E through H	Drywell	Harsh		L&N/8197-10-S
		TE-4328J through M				L&N/8197-10-S

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	2 Manufacturer/Model
MSIV-LCS pressure	2	PT-8404A through D	RB	Harsh	Type D variable; see Section II, Action Item 34	GE/555111DEAA4WCB
Primary safety relief valve position	2	PS-4400A through C	Drywell	Harsh	Type D variable; qualified in accordance with Section VII.L and Reference 3, App B, SCEW 303 (also see Section II, Action Item 38)	Pressure Controls/ A-17-1N
		PS-4401A through C	Drywell	Harsh		Pressure Controls/ A-17-1N
		PS-4402A through C	Drywell	Harsh		Pressure Controls/ A-17-1N
		PS-4403A through C	Drywell	Harsh		Pressure Controls/ A-17-1N
		PS-4404A through C	Drywell	Harsh		Pressure Controls/ A-17-1N
		PS-4405A through C	Drywell	Harsh		Pressure Controls/ A-17-1N
		PS-4406A through C	Drywell	Harsh		Pressure Controls/ A-17-1N
		PS-4407A through C	Drywell	Harsh		Pressure Controls/ A-17-1N
RCIC flow	2	FT-2509	RB	Mild	Type D variable; not required to be qualified for design basis LOCA (Reference 4)	
HPCI flow	2	FT-2309	RB	Mild	Type D variable; not required to be qualified for design basis LOCA (Reference 3, Section III)	
Core spray flow	2	FT-2130	NW crnr rm/1C-124	Harsh	Type D variable; see Section II, Action Item 35	GE/555-111BDAA3PDF
		FT-2110	SE crnr rm/1C-123	Harsh		GE/555-111BDAA3PDF

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	2 Manufacturer/Model
LPCI system flow (also RHR)	2	FT-1971A	SE crnr rm/1C-129A	Harsh	Type D variable; see Section II, Action Item 36	ITT-Barton/368
		FT-1971B	NW crnr rm/1C-129B	Harsh		ITT-Barton/368
RHR heat exchanger outlet temperature	2	TE-1945C	NW crnr rm	Harsh	Type D variable; see Section II, Action Item 37	Not available
		TE-1945E	SE crnr rm	Harsh		
RHR heat exchanger inlet temperature	2	TE-1945A	NW crnr rm	Mild	Type D variable for accidents other than design basis LOCA	
		TE-1945D	SE crnr rm	Mild		
		or				
		TE-4397A through H	Torus	Harsh	Type D variable for design basis LOCA	Not available
		TE-4398A through H (suppression pool water temp)	Torus	Harsh		Not available
Primary containment area radiation monitor	2	RE-9184A	Drywell	Harsh	Type E variable; implemented in accordance with NUREG 0737 (II.F.1-3)	Victoreen/877-1
		RE-9184B	Drywell	Harsh		Victoreen/877-1
		RE-9185A	Torus	Harsh		Victoreen/877-1
		RE-9185B	Torus	Harsh		Victoreen/877-1
Power supply status	2	Various	Control room	Mild	Type D variable; inherently Class 1E-powered	
Emergency damper position	2	ZS-5815A	SGT room	Harsh	Type D variable; see Section II, Action Item 39	Microswitch/OPD-AR
		ZS-5815B	SGT room	Harsh		Microswitch/OPD-AR
		ZS-5825A	SGT room	Harsh		Microswitch/OPD-AR
		ZS-5825B	SGT room	Harsh		Microswitch/OPD-AR
		ZS-7602A	SGT room	Harsh		Microswitch/OPD-AR
		ZS-7602B	SGT room	Harsh		Microswitch/OPD-AR
		ZS-7604U	A SGT room	Mild		Microswitch/OPD-AR
		ZS-7604V	A SGT room	Mild		Microswitch/OPD-AR

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	[2] Manufacturer/Model
Emergency damper position	2	ZS-7610A	RB-N	Mild		Microswitch/None
		ZS-7610B	RB-N	Mild		Microswitch/None
		ZS-7612A	RB-N	Mild		Microswitch/None
		ZS-7612B	RB-N	Mild		Microswitch/None
		ZS-7630A	RB-S	Mild		Microswitch/BZE6-2RN
		ZS-7630B	RB-S	Mild		Microswitch/BZE6-2RN
		ZS-7631A	RB-S	Mild		Microswitch/BZE6-2RN
		ZS-7631B	RB-S	Mild		Microswitch/BZE6-2RN
		ZS-7632A	RB-S	Mild		Microswitch/BZE6-2RN
		ZS-7632B	RB-S	Mild		Microswitch/BZE6-2RN
		ZS-7633A	RB-S	Mild		
		ZS-7633B	RB-S	Mild		
		ZS-7634A	RB-S	Mild		
		ZS-7634B	RB-S	Mild		
		ZS-7636A	RB-S	Mild		
		ZS-7636B	RB-S	Mild		
		ZS-7641A	Rad bldg	Mild		
		ZS-7641B	Rad bldg	Mild		
		ZS-6123A	Cont bldg	Mild		
		ZS-6123B	Cont bldg	Mild		
		ZS-7301A	Cont bldg	Mild		
		ZS-7301B	Cont bldg	Mild		
		ZS-7318A	Cont bldg	Mild		
		ZS-7318B	Cont bldg	Mild		
		ZS-6107A	Cont bldg	Mild		
		ZS-6107B	Cont bldg	Mild		

Variable Type Definitions (from BWR Owners Group Position on Regulatory Guide 1.97, Revision 2, dated May 1982)

Type A: Primary information on the basis of which operators take planned specified manually controlled actions.
Type B: Information about the accomplishment of plant safety functions.
Type C: Information about the breaching of barriers to fission product release.
Type D: Information about the operation of individual safety systems.
Type E: Information about the magnitude of the release of radioactive materials.

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	2 Manufacturer/Model

Notes

[1]KEY

Locations:

A SGT room	Above standby gas treatment room
Cont bldg	Control building
H&V CV rm	H&V control valve room
NE crnr rm	Northeast corner room
NW crnr rm	Northwest corner room
OGS	Off-gas stack
Rad bldg	Radwaste building
RB	Reactor building
RB-N	Reactor building, north
RB-S	Reactor building, south
RHR vlv rm	RHR valve room
RV	Reactor vessel
RWCU hx rm	Reactor water cleanup heat exchanger room
SE crnr rm	Southeast corner room
SGT room	Standby gas treatment room
St tunnel	Steam tunnel

[2]Consistent with the scope of this report, manufacturers and model numbers are provided for harsh environment equipment only.

TABLE 4-2

SAFETY DISPLAY INDICATION OF CONTAINMENT ISOLATION VALVES

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Penetration Number	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	Manufacturer/ Model[4]
Containment isolation valve position indication	1	X-7A	ZS-4412	Drywell	Harsh	See Ref 3 SCEW Sheet 264 for evaluation of EQ	NAMCo/EA-740
	1	X-7A	ZS-4413	St tunnel	Harsh	See Ref 3 SCEW Sheet 264 for evaluation of EQ	NAMCo/EA-740
	1	X-7B	ZS-4415	Drywell	Harsh	See Ref 3 SCEW Sheet 264 for evaluation of EQ	NAMCo/EA-740
	1	X-7B	ZS-4416	St tunnel	Harsh	See Ref 3 SCEW Sheet 264 for evaluation of EQ	NAMCo/EA-740
	1	X-7C	ZS-4418	Drywell	Harsh	See Ref 3 SCEW Sheet 264 for evaluation of EQ	NAMCo/EA-740
	1	X-7C	ZS-4419	St tunnel	Harsh	See Ref 3 SCEW Sheet 264 for evaluation of EQ	NAMCo/EA-740
	1	X-7D	ZS-4420	Drywell	Harsh	See Ref 3 SCEW Sheet 264 for evaluation of EQ	NAMCo/EA-740
	1	X-7D	ZS-4421	St tunnel	Harsh	See Ref 3 SCEW Sheet 264 for evaluation of EQ	NAMCo/EA-740
	1	X-8	MO-4423[2]	Drywell	Harsh	See Ref 3 SCEW Sheet 165 for evaluation of EQ	Limitorque/ SMB-00,AC Cl H
	1	X-8	MO-4424[2]	St tunnel	Harsh	See Ref 3 SCEW Sheet 207 for evaluation of EQ	Limitorque/ SMB-00,DC Cl B
	1	X-9A	MO-2312[2]	St tunnel	Harsh	See Ref 3 SCEW Sheet 187 for evaluation of EQ	Limitorque/ SMB-3,DC Cl B
	1	X-9A	MO-4441[2]	St tunnel	Harsh	See Ref 3 SCEW Sheet 190 for evaluation of EQ[3]	Limitorque/ SMB-4,AC Cl B
	1	X-9B	MO-2512[2]	Drywell	Harsh	See Ref 3 SCEW Sheet 207 for evaluation of EQ	Limitorque/ SMB-00,DC Cl B
	1	X-9B	MO-2740[2]	RB-S	Mild		
		X-9B	MO-4442[2]	St tunnel	Harsh	See Ref 3 SCEW Sheet 190 for evaluation of EQ[3]	Limitorque/ SMB-4,AC Cl B
	1	X-10	MO-2400[2]	Drywell	Harsh	See Ref 3 SCEW Sheet 165 for evaluation of EQ	Limitorque/ SMB-00,AC Cl H

Table 4-2 (continued)

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Penetration Number	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	Manufacturer/ Model[4]
Containment isolation valve position indication	1	X-10	MO-2401[2]	St tunnel	Harsh	See Ref 3 SCEW Sheet 207 for evaluation of EQ	Limitorque/ SMB-00,DC C1 B
	1	X-11	MO-2238[2]	Drywell	Harsh	See Ref 3 SCEW Sheet 215 for evaluation of EQ	Limitorque/ SMB-2,AC C1 H
	1	X-11	MO-2239[2]	St tunnel	Harsh	See Ref 3 SCEW Sheet 219 for evaluation of EQ	Limitorque/ SMB-3,DC C1 B
	1	X-12	MO-1908[2]	Drywell	Harsh	See Ref 3 SCEW Sheet 215 for evaluation of EQ	Limitorque/ SMB-2,AC C1 H
	1	X-12	MO-1909[2]	RHR vlv rm	Harsh	See Ref 3 SCEW Sheet 222 for evaluation of EQ	Limitorque/ SMB-2,DC C1 B
	1	X-13A	MO-2003[2]	RHR vlv rm	Harsh	See Ref 3 SCEW Sheet 193 for evaluation of EQ	Limitorque/ SMB-5,AC C1 B
	1	X-13A	MO-2004[2]	RHR vlv rm	Harsh	See Ref 3 SCEW Sheet 73 for evaluation of EQ	Electrodyne/ TN-24-400
	1	X-13B	MO-1904[2]	RHR vlv rm	Harsh	See Ref 3 SCEW Sheet 73 for evaluation of EQ	Electrodyne/ TN-24-400
	1	X-13B	MO-1905[2]	RHR vlv rm	Harsh	See Ref 3 SCEW Sheet 193 for evaluation of EQ	Limitorque/ SMB-5,AC C1 B
	1	X-15	MO-2700[2]	Drywell	Harsh	See Ref 3 SCEW Sheet 165 for evaluation of EQ	Limitorque/ SMB-00,AC C1 H
	1	X-15	MO-2701[2]	RWCU hx rm	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	X-16A	MO-2117[2]	RWCU hx rm	Harsh	See Ref 3 SCEW Sheet 229 for evaluation of EQ	Limitorque/ SMB-2,AC C1 B
	1	X-16A	MO-2115[2]	RB-S	Harsh	See Ref 3 SCEW Sheet 233 for evaluation of EQ	Limitorque/ SMB-0,AC C1 B
	1	X-16B	MO-2137[2]	RB-N	Harsh	See Ref 3 SCEW Sheet 229 for evaluation of EQ	Limitorque/ SMB-2,AC C1 B
	1	X-16B	MO-2135[2]	RB-N	Harsh	See Ref 3 SCEW Sheet 233 for evaluation of EQ	Limitorque/ SMB-0,AC C1 B
	1	X-17	MO-1900[2]	Drywell	Harsh	See Ref 3 SCEW Sheet 165 for evaluation of EQ	Limitorque/ SMB-00,AC C1 H
	1	X-17	MO-1901[2]	RB-N	Mild		
	1	X-19	ZS-3704	NE crnr rm	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-19	ZS-3705	NE crnr rm	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Pene-tration Number	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	Manufacturer/ Model[4]
Containment isolation valve position indication	1	X-22, N-229A	ZS-4371A	RB-SE	Mild		
	1	X-22, N-229A	ZS-4371C	RB-S	Mild		
	1	X-23A	ZS-5718A	Tosus room	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-23A	ZS-5719A	Torus room	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-23B	ZS-5718B	Torus room	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-23B	ZS-5719B	Torus room	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-24A	ZS-5704A	Torus room	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-24A	ZS-5703A	Torus room	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-24B	ZS-5704B	Torus room	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-24B	ZS-5703B	Torus room	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-25	ZS-4302	H&V CV rm	Mild		
	1	X-25	ZS-4303	H&V CV rm	Mild		
	1	X-25	ZS-4310	H&V CV rm	Mild		
	1	X-26	ZS-4312	RHR vlv rm	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-26, N220	ZS-4311	RHR vlv rm	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-26, N-220	ZS-4306	RHR vlv rm	Harsh	See Section II, Action Item 32	Microswitch/ OPD-AR
	1	X-26	ZS-4307	RHR vlv rm	Harsh	See Section II, Action Item 32	Microswitch/ OPD-AR
	1	X-32	ZS-1804A	RB-S	Mild		
	1	X-32	ZS-4378A,B	RB-S	Mild		
	1	X-32	ZS-1804B	RB-S	Mild		
	1	X-39A	MO-2000[2]	RB-S	Harsh	See Ref 3 SCEW Sheet 229 for evaluation of EQ	Limitorque/ SMB-2,AC CL B
	1	X-39A	MO-2001[2]	Tosus room	Harsh	See Ref 3 SCEW Sheet 184 for evaluation of EQ	Limitorque/ SMB-2,AC CL B

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Pene-tration Number	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	Manufacturer/ Model[4]
Containment isolation valve position indication	1	X-39A	ZS-4332A	RB-S	Harsh	See Section II, Action Item 32	Target Rock/ 72V-004
	1	X-39A	ZS-4332B	RB-S	Harsh	See Section II, Action Item 32	Target Rock/ 72V-004
	1	X-39A, N-211B	ZS-4334A	Tosus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-004
	1	X-39A, N-211B	ZS-4334B	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-004
	1	X-39B	MO-1902[2]	RHR vlv rm	Harsh	See Ref 3 SCEW Sheet 229 for evaluation of EQ	Limotorque/ SMB-2,AC Cl B
	1	X-39B	MO-1903[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 225 for evaluation of EQ	Limotorque/ SMB-2,AC Cl B
	1	X-39B	ZS-4331A	RHR vlv rm	Harsh	See Section II, Action Item 32	Target Rock/ 72V-004
	1	X-39B	ZS-4331B	RHR vlv rm	Harsh	See Section II, Action Item 32	Target Rock/ 72V-004
	1	X-39B, N-211A	ZS-4333A	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-004
	1	X-39B, N-211A	ZS-4333B	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-004
	1	X-41	ZS-4639	Drywell	Harsh	See Ref 3 SCEW Sheet 264 for evaluation of EQ	NAMCo/EA-740
	1	X-41	ZS-4640	RWCU hx rm	Harsh	See Section II, Action Item 32	NAMCo/SAI-131
	1	X-46F	SV-8105B	RB-N	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	X-46F	SV-8106B	RB-N	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	X-48	ZS-3728	Torus room	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-48	ZS-3729	Torus room	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	X-50B	SV-8101A	RB-S	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	X-50B	SV-8102A	RB-S	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	X-50D	SV-8105A	RB-S	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Penetration Number	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	Manufacturer/ Model[4]
Containment isolation valve position indication	1	X-50D	SV-8106A	RB-S	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	X-50E	SV-8103A	RB-S	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	X-50E	SV-8104A	RB-S	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	X-54	MO-4841A[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 169 for evaluation of EQ	Limitorque/ SMB-000,AC C1 B
	1	X-55	MO-4841B[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 169 for evaluation of EQ	Limitorque/ SMB-000,AC C1 B
	1	X-56C	SV-8101B	RB-N	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	X-56C	SV-8102B	RB-N	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	X-56D	SV-8103B	RB-N	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	X-56D	SV-8104B	RB-N	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	N-205	ZS-4300	NE crnr rm	Mild		
	1	N-205	ZS-4301	NE crnr rm	Mild		
	1	N-205	ZS-4309	NE crnr rm	Mild		
	1	N-210A, N-211A	MO-1932[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 157 for evaluation of EQ	Limitorque/ SMB-0,AC C1 B
	1	N-210A, N-211A	MO-1934[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 225 for evaluation of EQ	Limitorque/ SMB-2,AC C1 B
	1	N-210A	MO-2009[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 169 for evaluation of EQ	Limitorque/ SMB-000,AC C1 B
	1	N-210A	MO-2038[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 169 for evaluation of EQ	Limitorque/ SMB-000,AC C1 B
	1	N-210A	MO-2104[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 169 for evaluation of EQ	Limitorque/ SMB-000,AC C1 B
	1	N-210A	MO-2112[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 180 for evaluation of EQ	Limitorque/ SMB-1,AC C1 B
	1	N-210A	MO-2318[2]	HPCI room	Mild		
	1	N-210A	MO-2510[2]	RCIC room	Mild		
	1	N-210B, N-211B	MO-2005[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 157 for evaluation of EQ	Limitorque/ SMB-0,AC C1 B

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Penetration Number	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	Manufacturer/ Model[4]
Containment isolation valve position indication	1	N-210B, N-211B	MO-2007[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 225 for evaluation of EQ	Limitorque/ SMB-2,AC C1 B
	1	N-210B	MO-1935[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 169 for evaluation of EQ	Limitorque/ SMB-000,AC C1 B
	1	N-210B	MO-1970[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 169 for evaluation of EQ	Limitorque/ SMB-000,AC C1 B
	1	N-210B	MO-2132[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 180 for evaluation of EQ	Limitorque/ SMB-1,AC C1 B
	1	N-210B	MO-2124[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 169 for evaluation of EQ	Limitorque/ SMB-000,AC C1 B
	1	N-211A	MO-1933[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	N-211B	MO-2006[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	N-220	ZS-4308	RHR vlv rm	Harsh	See Section II, Action Item 32	Microswitch/ OPD-AR
	1	N-220	ZS-4313	RHR vlv rm	Harsh	See Section II, Action Item 32	Microswitch/ DTF2-2RN-RH
	1	N-229B	SV-8107A	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	N-229B	SV-8108A	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	N-229C	SV-8109A	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	N-229C	SV-8110A	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	N-229F	SV-8109B	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	N-229F	SV-8110B	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	N-229G	SV-8107B	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	N-229G	SV-8108B	Torus room	Harsh	See Section II, Action Item 32	Target Rock/ 72V-003
	1	N-224	MO-2516[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 211 for evaluation of EQ	Limitorque/ SMB-000,DC C1 B
	1	N-224	MO-2517[2]	RCIC room	Harsh	See Ref 3 SCEW Sheet 211 for evaluation of EQ	Limitorque/ SMB-000,DC C1 B

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Penetration Number	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	Manufacturer/ Model[4]
Containment isolation valve position indication	1	N-225A	MO-2069[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 157 for evaluation of EQ	Limitorque/ SMB-0,AC C1 B
	1	N-225A	MO-2012[2]	SE crnr rm	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	N-225A	MO-2015[2]	SE crnr rm	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	N-225B	MO-1913[2]	NW crnr rm	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	N-225B	MO-1921[2]	NW crnr rm	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	N-225B	MO-1989[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 157 for evaluation of EQ	Limitorque/ SMB-0,AC C1 B
	1	N-226	MO-2321[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 207 for evaluation of EQ	Limitorque/ SMB-00,DC C1 B
	1	N-226	MO-2322[2]	HPCI rm	Mild		
	1	N-227A	MO-2147[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	N-227A	MO-2100[2]	SE crnr rm	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	N-227B	MO-2146[2]	Torus room	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	M-227B	MO-2120[2]	NW crnr rm	Harsh	See Ref 3 SCEW Sheet 196 for evaluation of EQ	Limitorque/ SMB-00,AC C1 B
	1	N-231	ZS-4304	NE crnr rm	Harsh	See Section II, Action Item 32	Microswitch/ OPD-AR
	1	N-231	ZS-4305	NE crnr rm	Harsh	See Section II, Action Item 32	Microswitch/ OPD-AR

Notes

[1]KEY

Locations:

A SGT room	Above standby gas treatment room
Cont bldg	Control building
H&V CV rm	H&V control valve room

Post-Accident Monitoring Variable	Applicable Reg Guide 1.97, Rev 2, Design Qualification Category	DAEC Penetration Number	DAEC Plant Identification	DAEC Plant Location[1]	Post-Accident Environment Status	Comments	Manufacturer/ Model[4]

Locations (continued)

NE crnr rm	Northeast corner room
NW crnr rm	Northwest corner room
OGS	Off-gas stack
Rad bldg	Radwaste building
RB	Reactor building
RB-N	Reactor building, north
RB-S	Reactor building, south
RHR vlv rm	RHR valve room
RV	Reactor vessel
RWCU hx rm	Reactor water cleanup heat exchanger room
SE crnr rm	Southeast corner room
SGT room	Standby gas treatment room
St tunnel	Steam tunnel

[2]Position switch is contained within valve operator and does not have it's own unique plant identification number.

[3]These position switches are located inside Limitorque motor-actuated stop check valves in the feedwater system. The motor operators are not required to operate for the valves to perform their safety function, but the motor operators and associated position switches are qualified for the environmental conditions as evaluated on the referenced SCEW sheet for that model and type motor operator.

[4]Consistent with the scope of this report, manufacturers and model numbers are provided for harsh environment equipment only.