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DENTON,H. Office of Nuclear Reactor Regulation, Director

SUBJECT: Responds to 840507 request for addl info re environ qualification program.Justifications for continued operation contained in encl Sections II & X of Sept 1983 environ qualification rept.

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Iowa Electric Light and Power Company
May 31, 1984
NG-84-2213

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Response to NRC Request for Additional Information on DAEC Environmental Qualification Program

Dear Mr. Denton:

In response to the May 7, 1984 request for additional information sent to Iowa Electric by the NRC, the following information is provided:

1. The Justifications for Continued Operation (JCOs) for open action items are contained in Attachment 1. This is Section II of the DAEC Equipment Qualification Report dated September 1983, submitted to the NRC with Iowa Electric letter NG-83-3368 from R. W. McGaughy to H. Denton, dated September 30, 1983. As noted, the JCOs for open action items up to number 29 have been found technically acceptable in the Technical Evaluation Report (TER). The remaining JCOs are based on similar reasoning.

   For all JCOs associated with equipment that is assumed to fail, no significant degradation of any safety function or misleading information to the operator as a result of failure to equipment under the accident environment resulting from a design basis event will occur.

2. As described in Attachment 2 (Section X of the DAEC EQ Report), no distinction between safety-related and nonsafety-related equipment was made in identifying equipment to be included in the EQ program. This is consistent with 10 CFR 50.49(d). A list of harsh environment equipment essential for (or having potential for a failure affecting) the performance of a safety function following a design basis Loss Of Coolant Accident (LOCA) or High Energy Line Break Accident (HELBA) was generated. It is contained in the EQ Report, Appendix A. The LOCA and HELBA are the only design-basis accidents which result in significantly adverse environments to electrical equipment which is required for safe shutdown or accident mitigation. The list is based on reviews of all pertinent plant documents, primarily Piping and Instrumentation Diagrams and Electrical Schematics. A detailed description of methodology used in generating the list is provided in Attachment 2.

   The list of harsh environment electrical equipment essential for (or having potential for a failure affecting) the performance of a safety



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function following a design basis LOCA or HELBA includes the following:

a. Any auxiliary devices electrically connected directly into the control or power circuitry of safety-related equipment (e.g., automatic trips) whose failure due to postulated environmental conditions could prevent required operation of safety-related equipment.

b. Any electrical components in auxiliary systems (e.g., emergency service water, etc.) which are necessary for the required operation of safety-related equipment.

As discussed in Attachment 2, coordinated electrical fault protection (in the form of protective relays, circuit breakers, and fuses) is included in the plant design for electrical power distribution circuits. Therefore, harsh environment induced failures in nonsafety-related electrical equipment will be prevented from affecting other electrical loads. In addition, electrical circuits are physically separated as discussed in Chapter 8 of the DAEC UFSAR.

3. All design basis events in the scope of 10 CFR 50.49 which could potentially result in a harsh environment, including flooding outside containment, were addressed in identifying electrical equipment subject to environmental qualification. See the Technical Evaluation Report for the DAEC, TER-C5257-499, Section 4.3.5, for verification by Franklin Research Center of the technical adequacy of the flooding analysis outside containment.

This response, consisting of the foregoing letter and attachments is true and accurate to the best of my knowledge and belief.

IOWA ELECTRIC LIGHT AND POWER COMPANY

BY Richard W McGaugh
Richard W. McGaugh
Manager, Nuclear Division

Subscribed and sworn to Before Me on this 31st day of May 1984.

Kathleen M. Furman
Notary Public in and for the State of Iowa

RWM/BWR/dmb*

Attachments: 1. Section II of September, 1983 DAEC EQ Report
2. Section X of September, 1983 DAEC EQ Report

cc: B. Reid
L. Liu
S. Tuthill
M. Thadani
NRC Resident Office
Commitment Control Nos. 84-0103, 84-0114

## II. ACTION PLAN

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

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

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

1. ASCo SOLENOID VALVES

   Action complete (see Reference 2).

2. MAIN STEAM SAFETY RELIEF SOLENOID VALVES

   Action complete (see Reference 2).

3. TERMINAL BLOCKS

   Action complete (see Reference 2).

4. SOLENOID VALVE SV-4310

   Action complete (see Reference 3).

5. MAIN STEAM ISOLATION VALVE POSITION SWITCHES

   Action complete (see Reference 2).

6. PRESSURE TRANSMITTERS PDT-2046, -1947

a. Action Item Description

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

b. Resolution

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

c. Justification for Continued Operation

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

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

Action complete (see Reference 4)

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

Action complete (see Reference 4)

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

Action complete (see Reference 4)

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

a. Action Item Description

These flowmeters monitor main steam line leakage flow. They are S.K. Instrument Model 20-9651-8550, but were supplied by General Electric. These flowmeters provided a signal to their respective flow switches which cause system isolation on high flow. The flow switches are located in the control room (mild environment). The flowmeters and associated transmitter circuitry are located in the steam tunnel and must be qualified for a total dose of $2.1 \times 10^7$ 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 $1 \times 10^6$ rads. Additional qualification documentation is required.

b. Resolution

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons:

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

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

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

Action complete (see Reference 3).

12. EXHAUST UNITS 1K-25A,B

Action complete (see Reference 5).

13. MOTOR CONTROL CENTER 1D41

Action complete (see Reference 3).

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

a. Action Item Description

The motors for these air cooling units are Westinghouse with TEFC type enclosures. These units func[illegible] to provide post-accident room cooling f[illegible] RHR rooms. These units are located in [illegible]R rooms and must be qualified for a total [illegible] f 5.9x $10^6$ rads. Because of the equipment [illegible]ost-accident safety function and its location, [illegible]ualification for pressure, temperature, and humidity is not required.

Action Complete

b. Resolution

Subsequent investigation and evaluation indicate that it is not feasible to document qualification of these motors. They will [illegible] replaced during the Cycle 8 refueling outa[illegible]th new motors qualified to NUREG 0588, [illegible]y I. Installation will be via D[illegible].

Action Complete

c. Justification for Continued O[illegible]ion

Continued station operation is justified for the following reasons.

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

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

Action Complete

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.

Action Complete

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

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

Action complete (see Reference 4)

16. MOTOR-OPERATED VALVES

Action complete (see Reference 3).

17. FLOW SWITCHES FIS-2111, 2131

Action complete (see Reference 5).

18. PRESSURE SWITCHES PDIS-1971A,B

Action complete (see Reference 5).

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

a. Action Item Description

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

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

b. Resolution

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

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

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

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

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

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

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

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

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

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

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

3) Additional justification is provided below.

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

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

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

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

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

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

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

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

a. Action Item Description

Action Complete

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

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

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

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

Action Complete

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

b. Resolution

The electrical penetrations are shielded from the effects of water spray by a metal enclosure that totally surrounds the penetration and electrical termination area. The enclosure's orientation and adequacy f shielding against water spray was evaluat he adequate as a result of a walkdown during t e 7 refueling outage.

Action Complete

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

c. Justification for Continued Operation

Continued station operation is justified because the subject penetrations have been tested and environmentally qualified for severe LOCA environmental conditions, including high humidity and superheated steam. The penetration assemblies are inherently designed to prevent the intrusion of moisture into critical components. Because of its protection from direct spray effects, the splice materials do not require continued mechanical strength to perform its safety fun on of electrical insulation during a LC When not in tension or under continued stress, 's ability to withstand aging and radiation de increased. Also, potential for sigh t aging degradation of nylon will not occur un l 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.

Action Complete

21. AUTOMATIC VALVE COMPANY SOLENOID VALVE MODEL C5450-5

Action complete (see Reference 4)

22. FENWAL CONTROL UNITS MODEL 35003-0

a. Action Item Description

Control unit 5A1, A2, B1, B2, and CU-5837A1, A2, B1, B2 (Fen el 35003-0) are located in the standby gas t t system (SGTS) room at elevation 786' in reactor building. They are required to function as part of the SGTS. The

Action Complete

control units must be qualified for a 30-day integrated radiation dose of 1.6 x $10^8$ 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 external cable has determined its acceptability for at least 4.8 x $10^8$ rads and a 40-year qualified life. Qualification documentation is not available for the electronic switch.

Action Complete

b. Resolution

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

23. ELECTRODYNE VALVE OPERATOR MODEL TN-24-400

a. Action Item Description

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

b. Resolution

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

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

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

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

a. Action Item Description

The following solenoid valves (ASCo Models HT831665, 831665, 8320A6, and HB8302C25RU) are located in various area t the DAEC and are required to perform s unctions in several different systems.

Action Complete

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

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

b. Resolution

These valves were r[illegible]ced during the Cycle 7 refueling outage v[illegible] 1109. The Model 831665 and HT831665 valves [illegible]eplaced by ASCo Model NP831665E; the [illegible]320A6 was replaced by ASCo Model NP8320A173E[illegible] the Model HB8302C25RU was replaced by ASCo Model 206-832-2U. All these replacement models are qualified for a radiation dose in excess of 1.6 x $10^8$ rads and to the requirements of NUREG 0588, Category I.

Action Complete

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 x $10^7$ rads. This required radiation dose accounts for distance attenuation from the primary radiation service within the room. Because of the equipment's post-accident safety function and its location, qualification for pressure, temperature, and humidity is not required.

b. Resolution

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

26. GULTON INDUSTRIES COMPANY TEMPERATURE ELEMENT MODEL TCA-0646

Action complete (see Reference 4)

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

Action complete (see Reference 5).

28. INDUSTRIAL ENGINEERING EQUIPMENT COMPANY HEATERS MODEL TFZCP15900

Action complete (see Reference 5).

29. ROSEMOUNT TEMPERATURE ELEMENTS MODEL 104MA23ABBB

a. <u>Action Item Description</u>

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

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

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

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

30. BARKSDALE PRESSURE SWITCHES MODEL P1H-M85SS-V

Action complete (see Reference 5).

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

a. Action Item Description

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

b. Resolution

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

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

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

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

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

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

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

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

a. Action Item Description

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

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

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

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

b. Resolution

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

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

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

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

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

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

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

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

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

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

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

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

a. <u>Action Item Description</u>

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

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

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

b. Resolution

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

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

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

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

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

a. Action Item Description

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

b. Resolution

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

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

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

a. Action Item Description

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

b. Resolution

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

c. Justification for Continued Operation

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

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

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

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

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

a. Action Item Description

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

b. Resolution

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

c. Justification for Continued Operation

Continued plant operation is justified for the following reasons.

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

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

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

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

37. TEMPERATURE ELEMENTS FOR RHR HEAT EXCHANGER OUTLET TEMPERATURE INDICATION

a. Action Item Description

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

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

b. Resolution

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

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

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

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

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

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

a. Action Item Description

Action Complete

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

| | |
|---|---|
| PS-4400A,B,C | PS-4404A,B,C |
| PS-4401A,B,C | PS-4405A,B,C |
| PS-4402A,B,C | PS-4406A,B,C |
| PS-4403A,B,C | PS-440[illegible] B,C |

Action Complete

These pressure switches are qualified for drywell design basis LOCA conditions with the exception of the direct effects of containment [illegible]. 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 [illegible]uacy of drywell structural devices performing a[illegible]ar function at the DAEC requires investigation.

Action Complete

b. Resolution

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

Pressure switches PS-4400[illegible]C; PS-4401A,B,C; and PS-4403A,B,C through PS-[illegible]B,C were found to be effectively shielded from [illegible] spray effects by their sealed overhead junc[illegible]xes. Pressure switches PS-4402A,B,C were [illegible] to be effectively shielded by a combination of [illegible]verhead sealed condulets, drywell HVAC ducting, and drywell structural components.

Action Complete

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

a. Action Item Description

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

b. Resolution

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

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

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

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

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

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

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

a. Action Item Description

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

b. Resolution

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

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

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

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

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

a. Action Item Description

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

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

b. Resolution

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

c. Justification for Continued Operation

Continued station operation is justified for the following reasons.

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

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

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

# X. DESCRIPTION OF METHODOLOGY USED TO ADDRESS NONSAFETY-RELATED ELECTRIC EQUIPMENT FAILURES

## A. INTRODUCTION

1. 10 CFR 50.49 includes in its scope nonsafety-related electric equipment whose failure under postulated harsh environment conditions could prevent satisfactory accomplishment of a safety function. The possibility of failure of nonsafety-related equipment in a manner detrimental to safety was evaluated previously by a combination of the following methods.

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

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

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

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

## B. REVIEW AND DEVELOPMENT OF THE LIST OF EQUIPMENT (SYSTEM COMPONENTS) REQUIRING ENVIRONMENTAL QUALIFICATION

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

1. Equipment essential to safety functions were determined based on Section 3 of the DOR guidelines. These functions are as follows.

a. Achieve reactor cold shutdown

b. Achieve containment isolation

c. Provide reactor core cooling

d. Provide containment heat removal

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

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

3. Piping and Instrumentation Diagrams for each of the systems identified in Section B.2 were reviewed to identify system components essential to the performance of the system's safety function or whose failure would affect performance of the system's safety function and are required to function to mitigate the consequences of a LOCA or HELB as defined in NRC IE Bulletin 79-01B.

4. The entire instrument loop associated with each instrument identified in Section B.3 was reviewed to identify other components whose function was essential or whose failure could adversely affect operation of the instrument loop.

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

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

C. RELEVANT DAEC PLANT DESIGN CRITERIA

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

1. The DAEC plant design complies with the intent of IEEE Standard 279-1971 as described in UFSAR Sections 6.5.3.3, 7.2.1.2.2, 7.2.1.2.3, 7.2.1.2.4, and 7.3.1.2.2.2. For example, reactor protection system inputs to annunciators, recorders, and the computer are arranged so no malfunction of the annunciating,