TECHNICAL EVALUATION REPORT

EQUIPMENT ENVIRONMENTAL QUALIFICATION

JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK NUCLEAR GENERATING STATION

NRC DOCKET NO. 50-219

NRC TAC NO. 42524

FRC PROJECT C5257

NRC CONTRACT NO. NRC-03-79-118

FRC TASK 195

Prepared by

Franklin Research Center
The Parkway at Twentieth Street
Philadelphia, PA 19103

FRC Group Leader: C. J. Crane

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

NRC Lead Engineer: J. Lombardo

April 20, 1981

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Subcontractors

ECI = Energy Consultants, Inc.

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W = WESTEC Services, Inc.

IDENTIFICATION OF PROPRIETARY INFORMATION

Some of the information in this technical evaluation report was obtained from manufacturer's proprietary test reports. All proprietary information used in this report has been deleted.

1. INTRODUCTION

1.1 PURPOSE OF THE EVALUATION

The purpose of this report is to evaluate qualification documentation of nuclear power plant safety-related electrical equipment in accordance with criteria established by the NRC and to identify (1) equipment for which qualification documentation is adequate, i.e., substantiates that the equipment is capable of performing its specified design basis safety function when it is exposed to a harsh environment and (2) equipment for which qualification documentation is deficient, i.e., does not give reasonable assurance that the equipment is capable of performing its specified safety function. Where practical, this report presents recommendations for actions to remedy deficiencies.

1.2 GENERIC ISSUE BACKGROUND

The NRC criteria for reviewing the safety of nuclear power generating stations include the requirement that the qualification of safety-related electrical equipment be substantiated by auditable documentation of the program that establishes the ability of the equipment to function as specified in the station design. This report is restricted to a technical evaluation of the equipment's ability to function in harsh environments resulting from design basis events (DBEs).

Qualification criteria applied during the licensing of older nuclear power plants have been modified over the years, and specific industry standards concerning qualification have been revised as the design of reactor systems has changed and as regulatory and operating experience has accumulated. Examples of such standards are IEEE Standards 279-71, 323-74, 383-74, 317-76, 334-74, 381-77, 382-80, and 627-80. NRC NUREG documents 0413 and 0588 have been developed to address this topic. In particular, NUREG-0588 (published for comment in December 1979) formally presented the NRC staff positions regarding selected areas of environmental qualification of safety-related electrical equipment in the resolution of General Technical Activity A-24, "Qualification of Class IE Safety Related Equipment." The positions documented therein are applicable tc plants that are or will be in the construction permit or operating license review process.

Although qualification standards and regulatory requirements have undergone considerable development, all of the currently operating nuclear power plants are required to comply with 10CFR50, Appendix A, General Design Criteria for Nuclear Power Plants, Section I, Criterion 4. This criterion states in part that "structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents."

In 1977, the NRC staff instituted the Systematic Evaluation Program (SEP) to determine the degree to which the older operating nuclear power plants deviated from current licensing criteria. The subject of electrical equipment environmental qualification (SEP Topic III-12) was selected for accelerated evaluation as part of this program. Seismic qualification of equipment was to be addressed as a separate SEP topic. In December 1977, the NRC issued a generic letter to all SEP plant licensees requesting that they initiate reviews to determine the adequacy of existing equipment qualification documentation.

Preliminary NRC review of licensee responses led to the preparation of NUREG-0458, an interim NRC assessment of the environmental qualification of electrical equipment. This document concluded that "no significant safety deficiencies requiring immediate remedial actions were identified." However, it was recommended that additional effort should be devoted to examining the installation and environmental qualification documentation of specific electrical equipment in all operating reactors.

On May 31, 1978, the NRC Office of Inspection and Enforcement issued IE Circular 78-08, "Environmental Qualification of Safety-Related Electrical Equipment at Nuclear Power Plants," which required all licensees of operating plants (except those included in the SEP program) to examine their installed safety-related electrical equipment and ensure appropriate qualification documentation for equipment function under postulated accident conditions. Subsequently, on February 8, 1979, the NRC Office of Inspection and Enforcement issued IE Bulletin 79-01, which was intended to raise the threshold of IE Circular 78-08 to the level of Bulletin, i.e., action requiring a licensee response. This Bulletin required a complete re-review of the environmental qualification of safety-related electrical equipment as described in IE Circular 78-08.

The review of the licensee responses indicated certain deficiencies in the scope of equipment addressed, definition of harsh environments, and adequacy of qualification documentation. It became apparent that generic criteria were needed to evaluate the electrical equipment environmental qualification for both SEP and non-SEP operating plants. Therefore, during the second half of 1979, the Division of Operating Reactors (DOR) of the NRC issued internally a document entitled "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" [6].* (The document is hereafter referred to as the "DOR Guidelines.") The document was prepared as a screening standard for reviewing all operating plants, including SEP plants. It was originally intended that the licensees evaluate their qualification documentation in accordance with the DOR Guidelines. However, initial NRC review of this documentation, which was compiled to support licensee submittals, revealed the need for obtaining independent evaluations and for accelerating the qualification review program.

In October 1979, the NRC awarded Franklin Research Center (FRC) a contract to provide assistance in the "Review and Evaluation of Licensing Actions for Operating Reactors," which included an assignment for review of

*For References, see Section 6. Note that reference numbers are not presented in sequential order.

equipment environmental qualification documentation under SEP Topic III-12. FRC was to review equipment environmental qualification documentation and to present the results in the form of a Technical Evaluation Report for the 11 oldest plants (included in the SEP review).

On January 14, 1980, the NRC Office of Inspection and Enforcement issued the DOR Guidelines and IE Bulletin 79-01B, which expanded the scope of IE Bulletin 79-01 and requested additional information on environmental qualification of safety-related electrical equipment at operating facilities, excluding the 11 facilities undergoing the SEP review. This Bulletin cited the DOR Guidelines as the criteria to be used in evaluating the adequacy of the safety-related electrical equipment qualification. The scope of the review was expanded to include high energy line breaks (inside and outside containment) in addition to equipment aging and submergence. The NRC advised the licensees that the criteria contained in the DOR Guidelines would be used in its review of licensee submittals; problems arising from this review would be resolved using NUREG-0538 as a guide.

In early February 1980, the NRC decided that Indian Point Units 2 and 3 and Zion Units 1 and 2 should be included within SEP Topic III-12 for the purpose of equipment environmental qualification review.

On February 21, 1980, the NRC and representatives of the SEP Plant Owners Group held an open meeting at NRC headquarters to discuss an accelerated review program in accordance with the DOR screening guidelines. Representatives of the Indian Point Units and Zion Station also attended this meeting. The NRC formally issued to all licensees represented at the meeting the DOR Guidelines document which included a second document, "Guidelines for Identification of That Safety Equipment of SEP Operating Reactors for Which Environmental Qualification Is To Be Addressed" [6], together with the request that the licensees review their plant systems and provide additional equipment environmental qualification information to the NRC on an accelerated schedule.

In April 1980, the NRC organizational structure was modified and the Equipment Qualification Branch was formed within the new Division of Engineering. Responsibility for reviewing the status of equipment qualification for all plants was assigned to this branch.

On May 27, 1980, the NRC issued Memorandum and Order CLI-80-21 [10], specifying that Licensees and applicants must meet the requirements set forth in the DOR Guidelines and NUREG-0588 regarding environmental qualification of safety-related electrical equipment in order to satisfy 10CFR50, Appendix A, General Design Criteria, Section I, Criterion 4. This Order also established that the Safety Evaluation Reports on this subject, to be prepared by the NRC staff, must be issued on February 1, 1981 and that all subsequent actions to be taken by licensees to achieve full compliance with the DOR Guidelines or NUREG-0588 must be completed no later than June 30, 1982.

1.3 SPECIFIC ISSUE BACKGROUND

In a letter dated December 23, 1977, the NRC requested that Jersey Central Power & Light Company (JCP&L) review the status of environmental qualification for the safety-related electrical equipment at the Oyster Creek Nuclear Generating Station. Information requested from JCP&L included identification of electrical equipment required to perform safety functions while subjected to design basis accident environments, definitions of environmental service conditions at equipment locations, and the status of environmental qualification. In addition, documentation pertaining to qualification was to be compiled and organized for review by NRC. In response to this request, JCP&L provided information via submittals transmitted by letters dated February 24 and December 10, 1978. On March 10-13, 1980, NRC and FRC representatives visited the Oyster Creek plant, inspected safety-related systems and components, and discussed the program's requirements with JCP&L representatives. JCP&L provided additional information in letters dated April 11 and May 7, 1980. NRC and FRC representatives held a subsequent meeting with JCP&L representatives on October 9, 1980. The electrical equipment requiring qualification (limited to that located within the primary containment), the plant's environmental service conditions, and the qualification documentation for the plant were identified at this meeting and in subsequent communications.

FRC issued a Draft Interim Technical Evaluation Report to NRC on October 24, 1980. Copies of the report were transmitted to JCP&L by the NRC.

On August 29 and September 19, 1980, NRC notified JCP&L that all supplemental information on equipment environmental qualification must be submitted by November 1, 1980. On October 28, 1980, the Licensee sent the NRC a completely revised and expanded submittal of qualification information.

1.4 SCOPE OF THE EVALUATION

Environmental qualification of safety-related electrical equipment was selected by the NRC for accelerated review. Therefore, the scope of this report is limited to equipment that must function to mitigate the consequences of a loss-of-coolant accident (LOCA) or high energy line break (HELB) and equipment whose environment is adversely affected by those events. Qualification aspects not included within the scope of this evaluation are:

- o seismic qualification
- o equipment protection against natural phenomena
- o equipment operational service conditions (e.g., vibration, voltage, and frequency deviations)
- o equipment located where it is subject to outdoor environments
- o equipment protection against fire hazards
- o equipment protection against missiles.

2. NRC CRITERIA FOR ENVIRONMENTAL QUALIFICATION

2.1 CRITERIA PROVIDED BY THE NRC

The DOR screening guidelines used by FRC to evaluate the electrical equipment environmental qualification programs were:

- "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" [6]
- "Guidelines for Identification of That Safety Equipment of SEP Operating Reactors for Which Environmental Qualification Is To Be Addressed" [6].

These guidelines were issued for implementation to all licensees by the NRC in February 1980.

2.2 STAFF POSITIONS AND SUPPLEMENTAL CRITERIA

The NRC identified the following staff positions and supplemental criteria to be used in conjunction with the referenced DOR screening guidelines.

2.2.1 SERVICE CONDITIONS INSIDE CONTAINMENT FOR A LOSS-OF-COOLANT ACCIDENT (DOR Guidelines Section 4.1)

For pressurized water reactors (PWRs), the DOR Guidelines state that the containment temperature and pressure conditions as a function of time should be based on the most recent NRC-approved service conditions specified in the Final Safety Analysis Report (FSAR) or other licensee documentation. In the specific case of pressure-suppression type containments, the following minimum high temperature conditions may be used: (1) boiling water reactor (BWR) drywells -- 340°F for 6 hours and (2) PWR ice condenser lower compartments -- 340°F for 3 hours. As stated in Supplement 2 to IE Bulletin 79-01B [8], "these values are a screening device, per the Guidelines, and can be used in lieu of a plant-specific profile, provided that expected pressure and humidity conditions as a function of time are accounted for."

Service conditions should bound those expected for coolant and steam line breaks inside containment with due consideration given to analytical uncertainties. The steam line break condition should include superheated conditions, the peak temperature, and subsequent temperature/pressure profiles as functions of time. If containment spray is to be used, the impact of the spray on required equipment should be assessed.

The adequacy of a plant-specific profile depends on the assumptions and design considerations at the time the profiles were developed. The DOR Guidelines and NUREG-0588 provide guidance and considerations required to determine if the calculated plant-specific temperature/pressure profiles encompass the LOCA and HELB accidents inside containment.

2.2.2 SUBMERGENCE
(DOR Guidelines Section 4.1, Subitem 3; and Section 4.3.2, Subitem 3)

Equipment submergence (inside or outside containment) should be addressed where the possibility exists that submergence of equipment may result from HELBs or other postulated occurrences. Supplement 2 to IE Bulletin 79-01B [8] provides the following additional criterion: If the equipment satisfies the guidance and other requirements of the DOR Guidelines or NUREG-0588 for the LOCA and HELB accidents, and the licensee demonstrates that its failure will not adversely affect any safety-related function or mislead the operator after submergence, the equipment can be considered exempt from the submergence portion of the qualification requirements.

2.2.3 EQUIPMENT LOCATED IN AREAS NORMALLY MAINTAINED AT ROOM CONDITIONS
(DOR Guidelines Section 4.3.3)

Supplement 2 of IE Bulletin 79-01B [8] permits deferment of the review of environmental qualification for all safety-related equipment items located in plant areas where the equipment is not exposed to the direct effects of a HELB or to nuclear radiation emanating from circulation of fluids containing radioactive substances. At the licensee's option, the review may be deferred until after February 1, 1981.

By June 30, 1982, all safety-related electrical equipment potentially exposed to a harsh environment in nuclear generating stations licensed to operate on or before June 30, 1982 shall be qualified to either the DOR Guidelines or NUREG-0588 (as applicable). Safety-related electrical equipment is that required to bring the plant to a cold shutdown condition and to mitigate the consequences of the accident. It is the responsibility of the licensee to evaluate the qualification of safety-related electrical equipment to function in environmental extremes not associated with accident conditions and to document it in a form that will be available for the NRC to audit. Qualification to assure functioning in mild environments must be completed by June 30, 1982.

2.2.4 SIMULATED SERVICE CONDITIONS AND TEST DURATION (DOR Guidelines Section 5.2.1)

The Guidelines require that the test chamber environment envelop the required service conditions for a time equal to the period from the initiation of the accident until the service conditions return to normal. Supplement 2 to IE Bulletin 79-01B [8] provides the following additional criterion: "Equipment designed to perform its safety-related function within a short time into an event must be qualified for a period of at least 1 hour in excess of the time assumed in the accident analysis. The staff has indicated that time is the most significant factor in terms of the margins required to provide an acceptable confidence level that a safety-related function will be completed. The 1-hour qualification requirement is based on the acceptance of a type test for a single unit and the spectrum of accidents (small and large breaks) bounded by the single test."

2.2.5 DEFERMENT OF QUALIFICATION REVIEW

Supplement 3 to IE Bulletin 79-01B [9] permits the submittal of qualification documentation regarding the TMI Action Plan equipment and the equipment required to achieve and maintain a cold shutdown condition to be delayed as follows:

- "Qualification information for installed TMI Action Plan equipment must be submitted by February 1, 1981.
- Qualification information for future TMI Action Plan equipment (ref. NUREG-0737, when issued), which requires NRC pre-implementation review, must be submitted with the pre-implementation review data.
- Qualification information for TMI Action Plan equipment currently under NRC review should be submitted as soon as possible.
- Qualification information for TMI Action Plan equipment not yet installed which does not require pre-implementation review should be submitted to NRC for review by the implementation date.
- The qualification information for equipment required to achieve and maintain a Cold Shutdown condition ... will be submitted not later than February 1, 1981."

2.2.6 TEST SEQUENCE
(DOR Guidelines Section 5.2.3)

Supplement 2 to IE Bulletin 79-01B [8] provides the following additional criteria:

"Sequential testing requirements are specified in NUREG-0588 and the DOR Guidelines. Licensees must follow the test requirements of the applicable document.

1. If the test has been completed without aging in sequence, justification for such a deviation must be submitted.
2. If testing of a given component has been scheduled but not initiated, the test sequence/program should be modified to include aging.
3. Test programs in progress should be evaluated regarding the ability to comply by incorporating aging in the proper sequence. These would then fall in the first or second category."

2.2.7 RADIATION
(DOR Guidelines Sections 4.1.2, 4.2.2, and 4.3.2, Subitem 2)

Supplement 2 to IE Bulletin 79-01B [8] provides the following additional criteria:

"Both the DOR Guidelines and NUREG-0588 are similar in that they provide the methods for determining the radiation source term when considering

LOCA events inside containment (100% noble gases/50% iodine/1% particulates). These methods consider the radiation source term resulting from an event which completely depressurizes the primary system and releases the source term inventory to the containment.

NUREG-0578 provides the radiation source term to be used for determining the qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of containment as a result of LOCA. This method considers a LOCA event in which the primary system may not depressurize and the source term inventory remains in the coolant.

NUREG-0588 also provides the radiation source term to be used for qualifying equipment following non-LOCA events both inside and outside containment (10% noble gases/10% iodine/0% particulates).

When developing radiation source terms for equipment qualification, the licensee must ensure consideration is given to those events which provide the most bounding conditions. The following table summarizes these considerations:

	LOCA	Non-LOCA HELB
Outside Containment	NUREG-0578 (100/50/1 in RCS)[*]	NUREG-0588 (10/10/0 in RCS)
Inside Containment	Larger of NUREG-0588 (100/50/1 in containment) or NUREG-0578 (100/50/1 in RCS)	NUREG-0588 (10/10/0 in RCS)

Gamma equivalents may be used when consideration of the contributions of beta exposure has been included in accordance with the guidance given in the DOR Guidelines and NUREG-0588. Cobalt 60 is one acceptable gamma radiation source for environmental qualification of safety-related equipment. Cesium 137 may also be used."

*The numbers in parentheses represent % noble gases/% iodine/% particulates. RCS means reactor coolant system.

3. METHODOLOGY USED BY FRC

The Licensee, Jersey Central Power and Light Company, listed an extensive number of safety-related electrical equipment items in various locations of the Oyster Creek Nuclear Generating Station in its submittals to the NRC. FRC analyzed the Licensee's list and grouped together all identical equipment items located within plant areas that are exposed to the same environmental service conditions. This analysis reduced the list to 73 different equipment items to be reviewed. In this report, the term "equipment item" refers to a specific type of electrical equipment, designated by manufacturer and model, which is representative of all identical equipment in a plant area exposed to the same environmental service conditions (e.g., Flow Transmitter, Fischer & Porter, Model 10B2496, located within containment). Appendix A contains the environmental service conditions for each location, Appendix B contains a tabulation of the equipment items and locations (the tabulation does not include equipment covered by the evaluation deferment described in Section 2.2.3 of this report), and Appendix C lists the plant systems identified by the Licensee and the NRC as being essential to safety.

Using the list of safety-related electrical equipment items,* FRC reviewed each item in relation to:

- o NRC DOR Guidelines, as modified by NRC staff interpretations
- o Licensee definition of harsh service environments (Appendix A)
- o results of plant visit and equipment inspection
- o qualification documentation
- o analysis and/or justification of qualification
- o Licensee-proposed remedies for qualification deficiencies
- o Licensee-stated position concerning system or component function.

*In this report, the term "safety-related electrical equipment" refers to the equipment defined by the two NRC Guidelines referenced in Section 2.1.

Topics not within the scope of FRC evaluation are:

- o completeness of the Licensee's listing of safety-related equipment
- o acceptability of Licensee-provided environmental service conditions.

The initial results of FRC's review of the equipment environmental documentation were issued to NRC as a Draft Interim Technical Evaluation Report (DITER) on October 24, 1980 [7]. Qualification data summary forms used to summarize salient data compiled from the various information sources were included in the DITER.

In developing the present final Technical Evaluation Report (TER), FRC used the DITER and the Licensee submittals [1,2,3,4,5]. This information was analyzed by FRC to determine:

- o what specific response was made to the FRC DITER
- o whether the Licensee made any changes to the initial submittal
- o what additional information was supplied (e.g., analysis, test report, or justification for qualification)
- o whether any changes were made in the environmental conditions
- o whether any equipment was added or deleted.

All information was reviewed by FRC for conformance to the NRC criteria referenced in Section 2 of this report. As requested by the NRC, all qualification information developed in the Equipment Environmental Qualification (EEQ) program was used by the FRC reviewers, whether referenced by the Licensee or not. The qualification data summary forms were updated as appropriate and were then used to identify deviations from NRC criteria and the Licensee's qualification program. The final TER text was written primarily to address these deviations from the criteria. Items or test results not specifically cited by FRC implicitly satisfy the qualification criteria.

Upon completion of the final review for each equipment item, FRC developed an overall evaluation of the component and a specific conclusion with respect to its qualification. At the NRC's request, recommendations were made to resolve questions of deficient qualification. Based on the FRC conclusion, each equipment item was assigned to one of the generic qualification categories provided by the NRC. The NRC category descriptions follow.

NRC CATEGORIES AND DEFINITIONS

- NRC Category I.a
  EQUIPMENT THAT SATISFIES ALL APPLICABLE REQUIREMENTS OF THE DOR GUIDELINES

This category includes equipment items which are fully acceptable on the basis that all applicable criteria defined in the DOR Guidelines are satisfied and the equipment has been found to be qualified for the life of the plant.

- NRC Category I.b
  EQUIPMENT WITH ACCEPTABLE DEVIATIONS FROM THE DOR GUIDELINES

This category includes equipment items which do not satisfy one or more of the applicable criteria defined in the DOR Guidelines; however, sufficient information has been presented to determine that the specific deviations are acceptable and the equipment has been found to be qualified for the life of the plant.

- NRC Category II.a
  EQUIPMENT THAT SATISFIES ALL APPLICABLE REQUIREMENTS OF THE DOR GUIDELINES WITH THE EXCEPTION OF QUALIFIED LIFE

This category includes equipment items that are acceptable on the basis that all applicable criteria defined in the DOR Guidelines are satisfied with the exception of the qualified life criterion. With respect to qualified life, the equipment items have been found to have a qualified life which (1) is limited to a time interval less than plant life, (2) has not been adequately established in terms of calendar time, or (3) has not been evaluated by the licensee.

- NRC Category II.b
  EQUIPMENT THAT SATISFIES ALL APPLICABLE REQUIREMENTS OF THE DOR GUIDELINES WITH THE EXCEPTION OF QUALIFIED LIFE

This category includes equipment items which will be acceptable and will satisfy all applicable criteria defined in the DOR Guidelines with the exception of qualified life, provided that specific modifications are made on or before the designated date. When the modifications are complete, the equipment can be considered qualified with the exception of the qualified life criterion. With respect to qualified life, the equipment items have been found to have a qualified life which (1) is limited to a time interval less than plant life, (2) has not been adequately established in terms of calendar time, or (3) has not been evaluated by the licensee.

o NRC Category II.c
EQUIPMENT FOR WHICH DEVIATIONS FROM THE DOR GUIDELINES ARE JUDGED ACCEPTABLE WITH THE EXCEPTION OF QUALIFIED LIFE

This category includes equipment items which do not satisfy one or more of the applicable criteria defined in the DOR Guidelines; however, either (1) sufficient bases have been presented to allow a determination that the specific deviations are judged to be acceptable with the exception of qualified life criterion, or (2) the specific deviations are judged to be acceptable with the exception of qualified life criterion, based on review of the applicable qualification documentation associated with the overall equipment environmental qualification program. With respect to qualified life, the equipment items have been found to have a qualified life which (1) is limited to a time interval less than plant life, (2) has not been adequately established in terms of calendar time, or (3) has not been evaluated by the licensee.

o NRC Category III
EQUIPMENT THAT IS EXEMPT FROM QUALIFICATION

This category includes equipment items which are exempt from qualification on the basis that (1) the equipment does not provide a safety function (i.e., should not have been included in the equipment list submitted by the licensee), or (2) the specific safety-related function of the equipment can be accomplished by some other designated component which is fully qualified. In addition, any failure of the exempt equipment must not degrade the ability of qualified equipment to perform its required safety-related function.

o NRC Category IV.a
EQUIPMENT THAT HAS QUALIFICATION TESTING SCHEDULED BUT NOT COMPLETED

The qualification of equipment items in this category has been judged deficient or inadequate based upon review of the documentation provided by the licensee. However, the licensee has stated that the equipment item is scheduled to be tested by a designated date. The results of the testing will dictate the specific qualification category of the equipment item.

o NRC Category IV.b
EQUIPMENT FOR WHICH QUALIFICATION DOCUMENTATION IN ACCORDANCE WITH THE GUIDELINES HAS NOT BEEN ESTABLISHED

The qualification of equipment items in this category is deficient or inconclusive based upon review of the documentation provided by the licensee. This equipment is judged to have a high likelihood of operability for the specified environmental service conditions; however, complete and auditable records reflecting comprehensive qualification documentation have not been made available for review.

- NRC Category V
  EQUIPMENT THAT IS UNQUALIFIED

The DOR Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology and program be referenced and made available for review of all Class IE equipment.

The qualification of equipment items in this category has been judged to be deficient or inadequate, based upon review of the documentation provided by the licensee. The extent to which the equipment items fail to satisfy the criteria of the DOR Guidelines can be categorized as follows: (1) documentation reflecting qualification as specified in the DOR Guidelines has not been made available for review, (2) the documentation is inadequate, or (3) the documentation indicates that the equipment item has not passed the required tests.

- NRC Category VI
  EQUIPMENT FOR WHICH QUALIFICATION IS DEFERRED

This category includes equipment items which have been addressed by the licensee in the equipment environmental qualification submittals; however, the qualification review of this equipment has been deferred by the NRC in accordance with criteria presented in Sections 2.2.3 and 2.2.5 of this report.

4. TECHNICAL EVALUATION

General observations concerning the Licensee's approach to qualification are included in Section 4.1. Sections 4.2 through 4.7 identify the equipment items placed in each of the major NRC qualification categories in accordance with FRC's technical evaluation of the Licensee's documentation. The results of the evaluation are summarized in Section 4.8.

The technical evaluation of each equipment item is documented in the following format:

- o Original Text Taken From Draft Interim Technical Evaluation Report
- o Licensee Response
- o FRC Evaluation
- o FRC Conclusion.

All equipment item* numbers are associated with Reference 1.

4.1 METHODOLOGY USED BY THE LICENSEE

The final submittal of electrical equipment qualification documentation from the Licensee [1] was well organized and addressed the basic qualification requirements by means of system, equipment, and environmental analysis techniques. An FRC review of the documentation provided by the Licensee has generated the following observations.

*In this report, the term "equipment item" refers to a specific type of electrical equipment, designated by manufacturer and model, which is representative of all identical equipment in a plant area exposed to the same environmental service conditions (e.g., Flow Transmitter, Fischer & Porter, Model 10B2496, located within containment).

4.1.1 COMPLETENESS OF EQUIPMENT LIST

In the final submittal, the Licensee provided information for a large number (approximately 200) of equipment items. (The previous submittal [3] considered only equipment in the drywell of the primary containment.) The Licensee's equipment item list included only those safety-related electrical equipment items that are (i) installed in potentially "harsh" areas and (ii) needed for hot shutdown. The Licensee has elected to defer the review of equipment installed in "mild environments" and items needed for cold shutdown until after February 1, 1981, as discussed in Sections 2.2.3 and 2.2.5, and is continuing to assemble and review qualification information for these equipment items.

In Reference 1, the Licensee presented System Component Evaluation Work (SCEW) sheets for each safety-related equipment item for which the review is not deferred. These sheets summarize the pertinent environmental service conditions and identify available documentation references. FRC has analyzed the information in the SCEW sheets and has compiled a list of 73 equipment item groupings (henceforth referred to as "equipment items") for review in this Technical Evaluation Report. These equipment items consist of identical units having similar operational requirements and exposed to similar environmental conditions.

Discussions with the Licensee have indicated that motor control centers and possibly some switchgear have been overlooked as safety equipment located in "harsh" areas and required for hot shutdown. The Licensee stated that a revision to its most recent submittal [1] would be transmitted to rectify the oversight. The Licensee should also investigate the torus vacuum relief valve system to determine whether the vacuum relief valve solenoid and the differential pressure transmitter should be qualified. In addition, the Licensee should verify that no safety-related connectors or terminal blocks are located outside of the containment drywell.

4.1.2 ENVIRONMENTAL SERVICE CONDITIONS

4.1.2.1 TEMPERATURE AND PRESSURE PROFILES FOR THE CONTAINMENT DRYWELL

The Licensee states:

> The Oyster Creek containment temperature and pressure profile to be used for the environmental qualification of electrical equipment inside containment is derived from the most severe MSL break response with heat sinks and containment spray considered. This is the 0.75 ft^2 MSL break analysis. The results for this case are repeated in Figure 7-1 [Figure A-3 in Appendix A]. This plant-specific analysis represents a significant reduction from the 340°F for 6 hours recommended in NUREG-0588. The major reasons for the departure from the NUREG-0588 generic profile are the consideration of containment heat sinks and the initiation of containment spray.

The Licensee submittal indicates that Oyster Creek Station has automatic/ manual and redundant drywell containment sprays that can provide the long-term drywell heat sink and reduce the drywell temperature and pressure. The Licensee's drywell analysis used conservative energy release data and heat removal parameters, together with the assumption that the spray would be initiated 10 minutes after the break occurs. The NRC has reviewed the analysis and concurs that the MSLB accident analysis sets the limiting drywell service condition [11].

4.1.2.2 TEMPERATURE CONDITIONS IN THE REACTOR BUILDING

The Licensee has conducted extensive analysis to determine the environmental service conditions to which the safety-related electrical equipment needed for hot shutdown would be exposed in the event of postulated MSLB and HELB accidents. For areas where this equipment was located and the temperatures will exceed 100°F, temperatures as functions of time were presented in graphical form, and the peak temperature, pressure, and radiation levels were listed in Table 1. This information is included in Appendix A of this report.

4.1.2.3 RADIATION DOSE

The Licensee provided a description of methods for calculation and evaluation of dose values. Key statements from Reference 1 on the methodology are quoted below:

Analytical Methodology

EDS has calculated post-accident radiation exposures to vital equipment located inside the Oyster Creek containment due to airborne contamination and reactor vessel streaming. In addition, the radiation exposure contribution due to the station's normal forty-year operation has been considered. These calculations were performed by using the computer program QAD-P5A ... and simplified manual techniques. In all instances, the accident case source terms provided by JCP&L were utilized.

In order to calculate radiation exposures inside containment due to reactor vessel streaming, a one-hour post-accident source term composed of one hundred percent each of the noble gases, halogens, and the remainder isotopes was calculated using the source term data supplied by JCP&L. This source term was distributed within the region defined by the active volume of the fuel resulting in the reactor vessel source model input into the computer program QAD-P5A. Appropriate shielding credit was taken for the reactor vessel wall, the coolant within the reactor vessel, the self-shielding afforded within the fuel region, and the biological shield wall.

The calculated exposure rates outside the biological shield were held constant for forty years to determine the normal operation lifetime exposure. One-year post-accident integrated exposures were determined by applying an integration factor that accounted for the fission product radioactive decay during the one-year period following the accident.

Analysis Results and Discussion

Results of the inside containment exposure calculations due to reactor vessel streaming are shown in Table No. 1 for several locations and two electrical connector penetration lead shield thicknesses. The values shown indicate the normal operation forty-year lifetime exposure, the one-year post-accident integrated exposure, and the total. It should be pointed out that the reactor vessel streaming exposures presented here are for containment locations external to the biological shield. Exposures inboard of the biological shield would be considerably greater.

Table No. 2 indicates the results of calculations performed to determine radiation exposures inside the containment due to post-accident airborne activity. Values are presented for both the electrical connector penetration area and for a point midway between the outer biological shield wall and inner drywell wall. No credit has been taken for the lead end shields supplied with the penetrations as discussed in Section 3.0.

FRC agrees with the methodology applied by the Licensee. However, it appears that only the gamma contributions were listed in the tables. For beta-sensitive items such as electrical cables and perhaps some other items, the sum of the gamma plus beta dose should have been provided and identified.

4.1.2.4 "MILD AREA" ASSUMPTIONS FOR REACTOR BUILDING AND TURBINE BUILDING

The plant's environmental study of MSLB/HELB occurrences outside the containment identified several areas in the reactor building and turbine building which during an accident would not experience any change in temperature and pressure from the normal ambient conditions. From FRC's review of Reference 1, it was not clear whether the HVAC systems were assumed to be operational in order for the environmental service conditions of pressure and temperature to remain essentially normal in several areas. If the HVAC systems were assumed to be operational, then they are required by the DOR Guidelines to be redundant and powered from emergency electrical power systems. FRC has not had the opportunity to determine if redundant HVAC systems are available. The Licensee should either show that HVAC system operation was not assumed for the environmental calculations or provide evidence that the HVAC systems are redundant and fed by emergency power.

4.1.3 AGING AND QUALIFIED LIFE

The Licensee has not adequately addressed the related topics of aging and qualified life. The DOR Guidelines require that the Licensee:

- o establish (numerically) the qualified life for all equipment items containing components susceptible to degradation produced by heat and nuclear radiations
- o implement programs to review detailed surveillance and maintenance records to assure that equipment that exhibits age-related degradation is identified and replaced (or modified) as necessary.

Qualified life is the maximum time of normal service, under specified conditions, for which it can be demonstrated that the functional capability of the equipment at the end of the period is still adequate for it to perform its specified safety function(s) for applicable design basis events. The qualified life may be contingent on implementation of a specified maintenance program. It is acceptable for the qualified life of some subcomponents of an equipment item to be less than the qualified life of the item itself, provided a program for replacement of such subcomponents at intervals not exceeding their qualified lifetimes is specified and fulfilled. The qualified life of an equipment item may be changed during its installed life when justified by new information that permits a reanalysis of the qualification program.

Establishing the qualified life for equipment is a technically challenging task because of the paucity of information concerning the degradation of materials and components under the long-term exposure to the environmental service conditions in a nuclear power generating station. As is discussed more fully in Reference 13, with the possible exception of certain simple materials, there is no rigorous basis for establishing equipment qualified lifetimes for periods approaching an installed lifetime of 40 years. Furthermore, applicable information regarding possible long-term synergistic effects of temperature, humidity, nuclear radiations, etc. is extremely limited.

On virtually every SCEW sheet in Reference 1, the Licensee has stated (next to the parameter "Aging") a value of 40 years under both the "Specification" and "Qualification" headings. Presumably, these entries are intended as the qualified life.

In accordance with the Guidelines in this program, the licensees are required to establish a qualified life for equipment subject to thermal and radiation aging. In addition, surveillance, maintenance, and replacement programs should be established for equipment that may be subject to age-related degradation.

The licensees should review the qualified life values and the present installed life of the equipment in accordance with the DOR Guidelines to determine a replacement schedule for each equipment item (or subcomponents thereof). As noted above, these schedules may be revised as new information becomes available.

4.2 EQUIPMENT QUALIFIED FOR PLANT LIFE

This section includes equipment items which are fully acceptable on the basis that (1) all criteria defined in Section 2 of this report are satisfied or (2) sufficient data exist to determine that specific deviations are acceptable.

4.2.1 NRC Category I.a
EQUIPMENT THAT FULLY SATISFIES ALL APPLICABLE REQUIREMENTS OF THE DOR GUIDELINES

The equipment items in this section are fully acceptable on the basis that all applicable criteria defined in the DOR Guidelines are satisfied and the equipment has been found to be qualified for the life of the plant.

For the Oyster Creek Station, no equipment falls within this category.

4.2.2 NRC Category I.b
EQUIPMENT WITH ACCEPTABLE DEVIATIONS FROM THE DOR GUIDELINES

The equipment items in this section do not satisfy one or more of the applicable criteria defined in the DOR Guidelines; however, sufficient information has been presented to determine that the specific deviations are acceptable and the equipment has been found to be qualified for the life of the plant.

For the Oyster Creek Station, no equipment falls within this category.

4.3 EQUIPMENT QUALIFIED WITH RESTRICTIONS

This section includes equipment items that are acceptable on the basis that (1) all criteria defined in Section 2 of this report are satisfied with the exception of the qualified life criterion; (2) the equipment requires specific modification which, when completed, will establish full qualification with the exception of satisfying the qualified life criterion; or (3) with the exception of satisfying the qualified life criterion, deviations from the criteria presented in Section 2 have been found to be acceptable.

4.3.1 NRC Category II.a
EQUIPMENT THAT SATISFIES ALL APPLICABLE REQUIREMENTS OF THE DOR GUIDELINES WITH THE EXCEPTION OF QUALIFIED LIFE

The equipment items in this section are fully acceptable on the basis that all applicable criteria defined in the DOR Guidelines are satisfied with the exception of the qualified life criterion. With respect to qualified life, the equipment items have been found to have a qualified life which (1) is limited to a time interval less than plant life, (2) has not been adequately established in terms of calendar time, or (3) has not been evaluated by the Licensee.

4.3.1.1 Equipment Item No. 2
Solenoid Valves Located in the Reactor Building
ASCO Model NP-8344A70E
Drywell Vent and Purge Valves (V-26-16 and V-26-18)
(Licensee Reference 2.24)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These valves are ASCO Model NP-8344A70E and are qualified for LOCA environment. The results and description of the test are given in the ASCO Test Report No. AQS 21678/Tr, Revision A, dated March 1978.

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FRC EVALUATION:

FRC has reviewed Reference 2.24 and has the following comments:

1. During the qualification test program described in the reference,

   The test results must therefore be regarded as inconclusive until the uncertainties associated with the method of making the wiring interface with the solenoid, both in the plant and in the test, are resolved. The Guidelines state (Section 5.2.5):

   > "If a component fails at any time during the test, even in a so called 'fail safe' mode, the test should be considered inconclusive with regard to demonstrating the ability of the component to function for the entire period prior to the failure."

   They further state (Section 5.2.6):

   > "The equipment mounting and electrical or mechanical seals used during the type test should be representative of the actual installation for the test to be considered conclusive."

   However, because the environmental service conditions resulting from a HELB accident do not involve extremely high temperature, large radiation doses, or liquid spray, the deficiencies in the test are not of concern for this equipment item. The environmental parameters of the test program exceed by wide margins the plant-specific environmental service conditions stated by the Licensee. However, no justification for the Licensee's stated ambient temperature of only 77°F was given in the Reference 1 SCEW sheet. FRC notes that this value is lower than any other cited on the SCEW sheets, and does not correspond to HELB environmental conditions.

2. The pre-aging simulated in the test program was intended to represent an installed life (and hence a qualified life) of ambient temperature. The ambient temperatures at the installed locations within the plant are lower, and hence the qualified life is longer. The Licensee has not provided any justification for the claimed 40-year qualified life. An explicit, conservative determination of qualified life and replacement schedule (if needed) should be established.

FRC CONCLUSION:

This equipment is assigned to NRC Category II.a because a substantial period of qualified life and the ability to withstand the Licensee-stated HELB conditions at the installed location have been demonstrated. The Licensee should review the stated environmental conditions and establish a conservative qualified life. A surveillance program to monitor performance and identify any degradation requiring maintenance or replacement should also be implemented.

4.3.1.2 Equipment Item Nos. 49 and 50 (previously designated I6)
Electrical Cable Located Within the Drywell
49: General Electric Model S1-58145 Vulkene
50: General Electric Model S1-58073 Vulkene
(Original Licensee References 2.7, 2.11, 2.12, and 2.18;
Final Licensee References 2.16 and 2.21)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.5):

Reference 2.7 is discussed in Subsection 3.3.2.3 [4.5.2.17 in this report]. The connector tests described therein used cables removed from the plant. Reference 2.12 is a letter from General Electric stating that the Oyster Creek plant has two types of No. 12 AWG GE Vulkene cable installed "inside the containment" (FRC presumes this to mean within the drywell). This letter further states that the installed cable has an insulation thickness of 0.047 inch and that this is adequately represented by the No. 12 AWG GE Vulkene Type SIS cable included in the test program of the electrical penetrations conducted by GE in February 1975. The letter notes that the cables in the test program had an insulation thickness of 0.031 inch, and therefore the installed cable, having thicker insulation, "is considered qualified for the LOCA environment." The report of the penetration tests was not provided for review, so this reference must be regarded as irrelevant. Reference 2.18 is a report of a test performed on No. 12 AWG GE Vulkene cables removed from the Pilgrim Unit 1 plant and spliced. FRC comments are:

a. Although it appears that the tested samples are the same as the installed ones, complete documentation to substantiate this has not been provided. The Licensee should submit a listing of the type of cable (manufacturer, construction, materials) used for each item of Class 1E equipment within the drywell and provide complete documentation to relate this to valid test reports.

b. Neither test report included nuclear radiation exposures or consideration of aging. The thermal environmental parameters during the tests were adequate to represent plant-specific DBE conditions.

LICENSEE RESPONSE:

[No response provided.]

FRC EVALUATION:

The Licensee SCEW sheet identified the cable by specific type and added FIRL Report F-C4497-2 as evidence of qualification. FRC has reviewed the information provided by the Licensee, as well as the additional reference, and has the following comments:

1. The SCEW sheets 1-6A and 1-6B describe the cable installed in the drywell and relate it to the FIRL Report F-C4497-2 [2.21], resolving comment (a) of the DITER.

2. The cable tested in FIRL Report F-C4497-2 was pre-aged and irradiated to 200 Mrd, resolving comment (b) of the DITER.

FRC CONCLUSION:

This equipment is assigned to NRC Category II.a because qualification has been demonstrated by test, except for qualified life. The Licensee should establish a conservative qualified life (see Section 4.1.3).

4.3.1.3 Equipment Item No. 53 (previously designated I7)
Electrical Cable Located Within the Drywell
Rockbestos, Model Not Stated
(Final Licensee References 2.15 and 2.16)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.2.2):

Reference 2.15 is a manufacturer's qualification test report for three types of Rockbestos Firewall III cable (single conductor #16, #12, and #6 AWG). The first of these is stated to be instrumentation cable. The samples were thermally aged at 302°F for 1300 hours, which was intended to simulate 40 years "aging" in the plant at 194°F. The pre-aged cables were irradiated to 200 Mrd (gamma), and then exposed to a steam/chemical spray/moist atmosphere environment. Peak conditions were 346°F/113 psig steam for 3.6 hours; the total duration of the test was 140 days (30 days with steam plus 100 days at 200°F/100% RH). The cables were sprayed during the first 24 hours of the steam exposure with a solution of boric acid and sodium hydroxide. These conditions envelop the Licensee's expected MSLB and LOCA profiles by wide margins. The use of a different chemical solution in the spray is not regarded as a significant deficiency. Current and voltage loadings of the cable samples were applied during the 30-day steam exposure.

FRC concludes that this report establishes the environmental qualification of this equipment item according to the requirements of the Guidelines. This conclusion does not imply concurrence in the Licensee's implied claim that a 40-year qualified life has been established. The Arrhenius plot is based upon mechanical property data, and no information is presented to relate this to long-term electrical performance. The thermal aging exposure and the simulated LOCA exposure are both very severe, however. As a consequence, high confidence can be placed in the performance of the cable, and the qualified life can be expected to be quite long.

LICENSEE RESPONSE:

[No response provided.]

FRC EVALUATION:

As the Licensee provided no additional information, the original comments still apply.

FRC CONCLUSION:

This equipment is assigned to NRC Category II.a because qualification has been demonstrated by test except for qualified life. The Licensee should establish a conservative qualified life (see Section 4.1.3).

4.3.1.4 Equipment Item Nos. 31A and 32A
Solenoid Valves Located in the Steam Tunnel
31A: ASCO Model 206-832-3RU
32A: ASCO Model 206-301-3RU
MSIV Solenoid Valves and MSIV Position Indicators
(Final Licensee Reference 2.24)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

31A: NS-04A-L1, -L2, -L3
32A: NS-04B-L1, -L2

The MSIV solenoid valves are used to direct instrument air to hold open the outside containment main steam isolation valves. The MSIV position indication switches are utilized to provide a scram signal when the MSIVs are less than 90% open.

A loss of power or air to the MSIV solenoids causes the MSIVs to fail in the safe direction, closed. Also redundant protection is provided by the inside containment isolation valves that would not be affected by the environment created by outside containment breaks.

In the event the outside containment MSIV position switch did not provide a scram signal, two scram signals would still be available to ensure the reactor was shut down immediately for a MSLB. These two signals are the MSIV position switch signal from the inside valves and the reactor low water level signal, both of which would not be affected by the harsh environment created during this event.

The one-year integrated accident exposure of these components is at least two orders of magnitude below that which would cause any degradation.

Based upon the above discussion, it is expected that the main steam isolation function and reactor scram function required to mitigate MSLB outside containment will be accomplished.

FRC EVALUATION:

FRC has reviewed Reference 2.24 and has the following comments:

1. During the qualification test program described in the reference,

The test results must therefore be regarded as inconclusive until the uncertainties associated with the method of making the wiring interface with the solenoid, both in the plant and in the test, are resolved. The Guidelines state (Section 5.2.5):

> "If a component fails at any time during the test, even in a so called 'fail safe' mode, the test should be considered inconclusive with regard to demonstrating the ability of the component to function for the entire period prior to the failure."

They further state (Section 5.2.6):

> "The equipment mounting and electrical or mechanical seals used during the type test should be representative of the actual installation for the test to be considered conclusive."

However, because the environmental service conditions resulting from a HELB accident do not involve extremely high pressure, large radiation doses or liquid spray, the deficiencies in the test are not of concern for this equipment item. The environmental parameters of the test program exceed by wide margins the plant-specific environmental service conditions stated by the Licensee.

2. The pre-aging simulated in the test program was intended to represent an installed life (and hence a qualified life) of ambient temperature. The ambient temperatures at the installed locations within the plant are lower, and hence the qualified life is longer. An explicit, conservative determination of qualified life and a replacement schedule (if needed) should be established.

FRC CONCLUSION:

This equipment is assigned to NRC Category II.a because a substantial period of qualified life and the ability to withstand the Licensee-stated HELB conditions at the installed location have been demonstrated. The Licensee should review the stated environmental conditions and conservatively establish the qualified life (see Section 4.1.3).

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4.3.1.5 Equipment Item Nos. 4A and 34A
Motorized Valve Actuators Located in the Reactor Building
4A: Limitorque Model SMB-000
34A: Limitorque Model SMB-0
Spray and Cleanup Valves
(Final Licensee References 2.2, 2.3, 2.4, and 2.5)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

4A: Spray Valves (V-5-167 and V-5-147)

Following a worst-case line break (cleanup system line break outside drywell), these valves will remain in a non-harsh environment (95°F/16 psia). Further, these valves are not required to mitigate a cleanup system line break outside containment.

34A: Cleanup valves (V-16-2, -14, -61)

"Limitorque Qualified"

FRC EVALUATION:

1. Reference 2.2 is a letter from Limitorque stating that the test program in Reference 2.3 is applicable to this equipment item. However, with regard to Reference 2.4, Reference 2.2 states:

"Unfortunately, due to the date of supply, our records are not completely clear; however, we believe that our Qualification Report B0003 can be used to support the capability of the actuators to withstand irradiation."

2. Reference 2.3 is a report of a qualification test program conducted on a

The test program consisted of a 12-hour exposure to warm air saturated with water vapor (and). The performance of the actuator was monitored by cycling under load during the exposure (plus cycles before and after the exposure), and measuring the

exposure). Performance was satisfactory, but the

(There were no pre-aging, chemical spray, or nuclear radiation exposures in the test program.)

3. Limitorque Report B0003 [2.4] describes a qualification test program conducted on an SMB-0 MVA having a Reliance motor with Class B insulation, plus two additional motors. The MVA and motors were thermally aged and simultaneously operated (200 hours at 165°F; operation for 30 seconds in each direction once per hour for 176 hours), and the MVA was then operated for an additional 1817 cycles to simulate wear aging (the two extra motors were operated for an additional 15 minutes while the motors were unloaded). The MVA then received a nuclear radiation dose of 20 Mrd, and the motors 204 Mrd. Subsequently, the MVA and motors were seismically tested and subjected to a 16-day steam exposure test. Functional operation was demonstrated prior to and on five occasions during the latter exposure, the last immediately preceding the end of the test. Insulation resistance to ground was measured at each of these times. The MVA malfunctioned once (at 25.8 hours, just after the ambient temperature had been reduced from 250°F to 200°F). This malfunction was attributed to a "a momentary electrical short due to localized condensate buildup, a malfunction of the reversing contactor, or a combination of both." The IR readings decreased with time at each of the two temperature plateaus of the steam exposure, but ac current draw was not significantly affected.

The manufacturer concluded that "this test generically qualifies Limitorque Valve Actuators type SMB/SB for Class 1E Service outside primary containment for conditions as defined in this report." However, as noted in paragraph 1 above, Limitorque believes but cannot verify that this reference is applicable to the present evaluation.

4. Reference 2.5 is a letter from Limitorque that provides a general statement attempting to justify a 40-year qualified life based on the pre-aging exposures that were applied in the test programs. Because the applicability of Reference 2.4 is uncertain, and because there was no pre-aging in the test program reported in Reference 2.3, this letter appears irrelevant to the present evaluation. The Licensee should evaluate the susceptibility of the materials in the MVA to aging degradation and establish the conservative qualified life (refer to Section 4.1.3 for additional comments).

5. Because the environmental service conditions during an accident do not deviate appreciably from normal non-accident conditions, FRC considers that Reference 2.3 satisfies the Guidelines requirements, except for qualified life.

FRC CONCLUSION:

This equipment is assigned to NRC Category II.a because the Guidelines requirements are satisfied except for qualified life. The Licensee should establish a conservative qualified life (see Section 4.1.3).

4.3.2 NRC Category II.b
EQUIPMENT THAT SATISFIES ALL APPLICABLE REQUIREMENTS OF THE DOR GUIDELINES WITH THE EXCEPTION OF QUALIFIED LIFE PROVIDED THAT SPECIFIC MODIFICATIONS ARE MADE

The equipment items in this section will be acceptable and will satisfy all applicable criteria defined in the DOR Guidelines with the exception of qualified life provided that specific modifications are made on or before the designated date. When the modifications are complete, the equipment can be considered qualified with the exception of the qualified life criterion. With respect to qualified life, the equipment items have been found to have a qualified life which (1) is limited to a time interval less than plant life, (2) has not been adequately established in terms of calendar time, or (3) has not been evaluated by the Licensee.

For the Oyster Creek Station, no equipment falls within this category.

4.3.3 NRC Category II.c
EQUIPMENT FOR WHICH DEVIATIONS FROM THE DOR GUIDELINES ARE JUDGED ACCEPTABLE WITH THE EXCEPTION OF QUALIFIED LIFE

The equipment items in this section do not satisfy one or more of the applicable criteria defined in the DOR Guidelines; however, either (1) sufficient bases have been presented to allow a determination that the specific deviations are judged to be acceptable with the exception of the qualified life criterion, or (2) the specific deviations are judged to be acceptable with the exception of the qualified life criterion based on a review of the applicable qualification documentation associated with the overall equipment environmental qualification program. With respect to qualified life, the equipment items have been found to have a qualified life which (1) is limited to a time interval less than plant life, (2) has not been adequately established in terms of calendar time, or (3) has not been evaluated by the Licensee.

4.3.3.1 Equipment Item Nos. 3A, 3B, 4B, and 34B
Motorized Valve Actuators Located in the Reactor Building
3A and 3B: Limitorque Model SMB-00
Containment Spray Valves
4B: Limitorque Model SMB-000
Containment Spray Valves
34B: Limitorque Model SMB-0
Core Spray Valves
(Final Licensee References 2.2, 2.3, 2.4, and 2.5)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

3A: Containment Spray Valves (V-21-5 and V-21-11)

The peak temperatures seen by these valves are 140°F (V-21-5) and 250°F (V-21-11) following a cleanup system line break outside the drywell. However, these valves are not required to mitigate a line break outside the drywell. If a line break is inside the drywell, the valves located outside the drywell will not experience a high temperature or pressure. The valves will only see a rise in radiation level. However, these valves are normally open and will stay open even if the valve operator is de-energized. Therefore, the ability of the system to be used for drywell and torus cooling will not be affected.

3B: Containment Spray Valves (V-21-1, -3, -7, -9)

The peak temperature and pressure seen by these valves following a worst case line break (a MSLB outside drywell) will be 165°F and 15 psia.

4B: Spray Valves (V-21-13 and V-21-17)

Valve V-21-17 will not be affected by the break and thus will remain in the non-harsh environment (77°F and 15 psia). The other valve (V-21-13) will experience a peak temperature of 140°F. However, these valves are not required to mitigate a line break outside drywell. In any case, these valves are normally open and will stay open even if the valve

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operator fails to function. Therefore, the ability of the system to be used for drywell and torus cooling will not be affected.

34B: Shutdown Cooling Valves (V-17-1, -2, -3, -55, -56, -57)

"Limitorque Qualified"

FRC EVALUATION:

FRC has reviewed the references cited by the Licensee and has the following comments:

1. Reference 2.2 is a letter from Limitorque stating that the test program in Reference 2.3 is applicable to this equipment item. However, with regard to Reference 2.4, Reference 2.2 also states:

"Unfortunately, due to the date of supply, our records are not completely clear; however, we believe that our Qualification Report B0003 can be used to support the capability of the actuators to withstand irradiation."

2. Reference 2.3 is a report of a qualification test program conducted on a

The test program consisted of a 12-hour exposure to warm air saturated with water vapor (and). The performance of the actuator was monitored by cycling under load during the exposure (plus cycles before and after the exposure), and the

exposure). Performance was satisfactory, but the (There were no pre-aging, chemical spray, or nuclear radiation exposures in the test program.)

3. Limitorque Report B0003 [2.4] describes a qualification test program conducted on a SMB-0 MVA having a Reliance motor with Class B insulation, plus two additional motors. The MVA and motors were thermally aged and simultaneously operated (200 hours at 165°F; operation for 30 seconds in each direction once per hour for 176 hours), and then the MVA was then operated for an additional 1817 cycles to simulate wear aging (the two extra motors were operated for an additional 15 minutes while the motors were unloaded). The MVA then received a nuclear radiation dose of 20 Mrd, and the motors 204 Mrd. Subsequently, the MVA and motors were seismically tested and subjected to a

16-day steam exposure test. Functional operation was demonstrated prior to and on five occasions during the latter exposure, the last immediately preceding the end of the test. Insulation resistance to ground was measured at each of these times. The MVA malfunctioned once (at 25.8 hours, just after the ambient temperature had been reduced from 250°F to 200°F). This malfunction was attributed to a "a momentary electrical short due to localized condensate buildup, a malfunction of the reversing contactor, or a combination of both." The IR readings decreased with time at each of the two temperature plateaus of the steam exposure, but ac current draw was not significantly affected.

The manufacturer concluded that "this test generically qualifies Limitorque Valve Actuators type SMB/SB for Class 1E Service outside primary containment for conditions as defined in this report." However, as noted in paragraph 1 above, Limitorque believes but cannot verify that this reference is applicable.

4. Reference 2.5 is a letter from Limitorque that provides a general statement attempting to justify a 40-year qualified life based on the pre-aging exposures that are applied in the test programs. Because the applicability of Reference 2.4 is uncertain, and because there was no pre-aging in the test program reported in Reference 2.3, this letter appears to be irrelevant to the present evaluation. The Licensee should evaluate the susceptibility of the materials in the MVA to aging degradation and establish the conservative qualified life (refer to Section 4.1.3 for additional comments).

5. FRC considers that all Guidelines requirements except those pertaining to nuclear radiations and aging have been satisfied. Aging was discussed above. With regard to nuclear radiations, it appears that the dose levels are small enough that the Licensee should have no difficulty in establishing qualification by analysis.

FRC CONCLUSION:

This equipment is assigned to NRC Category II.c. Although complete qualification documentation has not been made available to demonstrate

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total compliance with the DOR Guidelines, it is expected that the Licensee will be able to demonstrate qualification for all environmental service conditions (including nuclear radiation exposure) and a significant period of qualified life (less than plant life).

4.3.3.2 Equipment Item No. 52 (previously designated I10)
Electrical Cable Located Within the Drywell
Kerite, Model Not Stated
(Final Licensee References 2.16 and 2.23)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.6):

Licensee Reference 2.23 is a test report covering three sets of samples: one for control cable (7/C No. 12 AWG) and two different constructions of power cable (1/C No. 6 AWG); plus eight splice samples. FRC's comments on this reference are as follows:

a. The test specimen must be the same as the equipment being qualified. The Licensee did not present an analysis comparing the impact of deviation between the test specimen's specific design features, materials (specifically, the formulations used in the insulation and jacket), and production procedure, and those of the cables installed in the plant. Therefore, the validity of the test as evidence for qualification has not been established.

b. The test program consisted of steam and boric acid spray exposures , plus cooldown simultaneous with exposure to gamma radiation. The total dose administered to various samples was either The samples were electrically loaded during the simulated LOCA exposures except for periods when electrical measurements were being made. The peak temperature and pressure in the test exceeded the plant-specific accident values, but the profile was not completely enveloped. Also, the chemical solution of the spray was different from that in plant. Because of the overall severity of the test, these deficiencies are judged to be minor and acceptable. The nuclear radiation exposure was more than adequate.

c. The cable samples were not thermally pre-aged prior to the simulated LOCA exposure, as is required by the Guidelines when it has not been shown that the materials are not subject to aging degradation.

it is particularly
It is also important
that (i) acceptance criteria be established for these cables,

considering their plant-specific application, and (ii) a determination be made that the electrical current loadings in the test are adequate. Also, the qualified life should be established.

LICENSEE RESPONSE:

[No response provided.]

FRC EVALUATION:

As a result of review of other test reports referenced by Licensee in the EEQ program for SEP plants, FRC has also reviewed FIRL Reports F-C4158, F-C4020-1, and F-C4040-2 (FIRL test on Kerite cable). The cables covered by these reports are:

F-C4518: 7/C No. 12 AWG

F-C4020-1: 7/C No. 12 AWG

F-C4020-2: 7/C No. 12 AWG

F-C4020-2: 1/C No. 6 AWG

For the cables in FIRL Report F-C4020-1, the insulation resistance was noticeably lower after thermal aging, then decreased by a factor of about 100 after irradiation, and by another factor of about 1000 the first 1.5 hour at 346°F/113 psig in the test chamber. The report states in the conclusion that the cables were able to maintain load ( ) for 2 days (1 cable), days (1 cable), and days (2 cables) after start of the specified LOCA.

For the cables reported in FIRL Report F-C4020-2, the temperature/pressure conditions of the steam exposure were rapid heating from psig to 346°F/113 psig, which was held for 3 hours, followed by cooldown to 140°F in 2 hours. a second rapid heating to 346°F/113 psig (held for 3 hours), and then a gradual stepwise drop in temperature to psig, which was held for days. These tests were conducted in 1975 and envelop the Oyster Creek conditions in Appendix A. For the cables reported in FIRL Report F-C4158, the temperature/pressure conditions were hours at , days at (with buffered boric acid spray), then days at and finally ambient. This test also envelops the specified conditions for the Oyster Creek Station. All three tests involved simultaneous nuclear radiation and steam/spray exposures.

The Licensee submittal does not state whether cables are exposed or in conduit. The tests reviewed involved exposed cables, and the radiation dose rate and total dose exceeded the Oyster Creek requirements. As discussed in Section 4.1.3, FRC does not agree with the manufacturer's claim of a lifetime in excess of 40 years.

FRC CONCLUSION:

This equipment is assigned to NRC Category II.c because FRC is aware of test results that qualify the cable. The Licensee should establish a conservative qualified life (see Section 4.1.3).

4.3.3.3 Equipment Item No. 54 (previously designated I11)
Electrical Splices Located Within the Drywell
Raychem Type WCSF
(Final Licensee References 2.9 and 2.16)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.7):

Licensee Reference 2.16 is a test report for Raychem splices. Although the test program reported may be adequate, the Licensee has not established that the splices in the plant used the materials and techniques covered in this test program. (The absence of the plant-specific chemical spray is not regarded as a serious deficiency.)

LICENSEE RESPONSE:

[The Licensee identified the splices as WCSF type, referenced an additional test report (Wyle No. 44114-2), and noted that the radiation level due to an accident is being re-estimated and is expected to be lower than the 57 Mrad used in the evaluation.]

FRC EVALUATION:

After reviewing documentation on splices referenced previously and information supplied by other Licensees for the EEQ program, FRC has the following comments:

a. According to information provided to various licensees by Raychem Corp., failure in the cable insulation may also result in failure of the splice.

b. Testing reports showed WCSF-type splices to be satisfactory with the Kerite, Rockbestos, and GE Vulkene cables identified in SCEW sheets for Oyster Creek, but tests for the Tensolite cable (Equipment Item No. 51) were not reported.

c. The testing conditions enveloped the pressure, temperature, and radiation levels applicable to Oyster Creek. Chemical sprays in the tests (boric acid solutions buffered to a pH of 9.5-10.5) differed from the Oyster Creek spray. However, as noted in the DITER above, the difference in spray is not considered a serious deficiency.

d. As discussed in Section 4.1.3, FRC does not agree with the manufacturer's stated 40-year life for this equipment. The Licensee should obtain information to establish a conservative qualified life for splices on all the cables installed in the Oyster Creek Station.

FRC CONCLUSION:

Except when used on Tensolite cables, these splices are assigned to NRC Category II.c because test reports supported compliance with all Guidelines criteria except qualified life. WCSF-type splices on Tensolite cables, if any, would be assigned to NRC Category IV.b because they are likely to be satisfactory but documentation is lacking. The Licensee should establish a conservative qualified life for each cable/splice system and a surveillance program to monitor performance and identify any degradation which would indicate the need for maintenance or replacement (see Section 4.1.3).

4.4 NRC Category III
EQUIPMENT THAT IS EXEMPT FROM QUALIFICATION

The equipment items in this section are exempt from qualification on the basis that (1) the equipment does not provide a safety function (i.e., should not have been included in the equipment list submitted by the Licensee), or (2) the specific safety-related function of the equipment can be accomplished by some other designated equipment that is fully qualified. In addition, any failure of the exempt equipment must not degrade the ability of qualified equipment to perform its required safety-related function.

4.4.1 Equipment Item No. 39
Electric Motors Located in the Reactor Building
General Electric Model 5K818841C45
Core Spray Booster Pumps (NZ-03-A through NZ-03-0)
(Final Licensee Reference 2.14)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

The purpose of the core spray booster pumps is to provide additional pressure increase to the core spray water discharged by the core spray pumps. This ensures rated core spray flow will be established at a reactor pressure of 110 psig. The core spray system consists of two independent systems, each of which can accomplish its safety function even considering a single active failure. Pumps A and C are in System I and B and D are in System II. The environmental conditions in the area of the B and D pumps are nonharsh when only temperature and pressure are considered. Therefore, there always will be at least one system available to carry out its safety function. The one-year integrated accident exposure to the pumps in System I is on the order of 1 Mrads, and System II pump exposure is on the order of 0.1 Mrads. By evaluation, it has been determined that there will be no detrimental radiation effects up to radiation exposures of 200 Mrads.

Based on the above considerations, it is expected that even considering the worst-case HELBs, there will be at least one core spray system booster pump available to deliver rate core spray flow to the reactor if that should be required.

(Qualified) Per GE report

FRC EVALUATION:

The Licensee has stated that the core spray booster pumps are each able to supply 100% of the core cooling needs. Accordingly, even if two of the pumps located in the harsh area of the reactor building were rendered inoperable by the MSLB and a single failure prevented one of the mild area pumps from operating, a 100% pump located in another area (described by the Licensee as mild) would still remain to furnish the necessary cooling to the core. A MSLB in the reactor building should be of short duration so that the remaining pump would not have to operate for more than a few hours or days to bring the plant to a safe shutdown. On this basis, the pump motor can be considered exempt from qualification.

FRC CONCLUSION:

The core spray booster pump motors are assigned to NRC Category III because there is sufficient redundancy with equipment located in a mild area to withstand a single failure and still provide the necessary system function capability.

4.5 EQUIPMENT FOR WHICH DOCUMENTATION CONTAINS DEVIATIONS FROM THE GUIDELINES THAT ARE JUDGED UNRESOLVED

This section includes equipment items which are deficient on the basis that all criteria defined in the DOR Guidelines are not satisfied. However, the equipment item is either scheduled to be tested or is judged to have a high likelihood of operability.

4.5.1 NRC Category IV.a
EQUIPMENT THAT HAS QUALIFICATION TESTING SCHEDULED BUT NOT COMPLETED

The qualification of the equipment items in this section has been judged deficient or inadequate based upon review of the documentation provided by the Licensee; however, the Licensee has stated that the equipment item is scheduled to be tested by a designated date. The results of the testing will dictate the specific qualification category he equipment item.

For the Oyster Cr"ek Station, no equipment falls within this category.

4.5.2 NRC Category IV.b
EQUIPMENT FOR WHICH QUALIFICATION DOCUMENTATION IN ACCORDANCE WITH THE GUIDELINES HAS NOT BEEN ESTABLISHED

The qualification of the equipment items in this category is deficient or inconclusive based upon review of the documentation provided by the Licensee. This equipment is judged to have a high likelihood of operability for the specified environmental service conditions; however, complete and auditable records reflecting comprehensive qualification documentation have not been made available for review.

4.5.2.1 Equipment Item No. 1
Pressure Switches Located in the Reactor Building
Dresser Model 1539 VX
Automatic Depressurization System (ADS) Pressure Switches (IA83A through IA83E)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

The ADS provides for a controlled blowdown of the reactor pressure vessel to rapidly reduce pressure during a small pipe break. This permits core spray actuation prior to uncovering the fuel. The pressure switches will open the electromatic relief valves in the ADS on an overpressure condition in the reactor pressure vessel. Each pressure switch is installed at a different location outside the Drywell and a single HELB in the vicinity will not subject all five switches to a peak temperature and pressure. These switches are necessary only for over-pressurization protection and their failure does not affect the ability of the Control Room operator to manually operate ADS valves in order to achieve a controlled cooldown. Even without the relief valves, reactor vessel overpressure protection is provided by 16 safety valves located within the containment. Therefore, they will be unaffected by any HELBs outside containment.

FRC EVALUATION:

The Licensee has not provided, and FRC has found no other sources of, valid qualification documentation for this equipment. Therefore, qualification has not been established in accordance with the requirements of the Guidelines. However, some of this equipment is likely to function adequately because its safety function is expected to be performed early in the accident scenario and not all of the pressure switches are expected to be exposed to a harsh environment at the same time.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because, although valid qualification documentation has not been provided, the Licensee has shown that the equipment is likely to function. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to a program of equipment qualification or replacement by June 1982.

4.5.2.2 Equipment Item No. 4C (previously designated IA-2)
Motorized Valve Actuators Located Within the Drywell
Limitorque Model SMB-000
Main Steam Line Isolation (V-1-106, 107)
(Original Licensee Reference 2.4; Final Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.1):

Reference 2.4 is a test report of a qualification test for an SMB-0 actuator. FRC has the following comments with regard to this reference.

a. The test report is for a Limitorque Model SMB-0 actuator with a Reliance motor having Class B insulation. The Guidelines require that the test specimens be the same as the equipment being qualified. The Licensee did not present an analysis comparing the impact of deviations between the test specimen's specific design features, materials, and production procedures and those of the installed equipment. Therefore, an independent conclusion can not be reached regarding the extent to which the tested equipment is similar to that installed in the plant, and the validity of the test, as evidence of qualification, has not been established.

b. The test program included wear/thermal/humidity/seismic aging, vibration test to simulate severe seismic events, and a steam exposure. The environmental parameters, aging considerations, and other aspects of the test program were intended to demonstrate qualification of equipment located outside of the primary containment; they are not adequate to demonstrate qualification for these equipment items located within the containment drywell.

The Licensee has stated that these equipment items will be replaced with fully qualified equipment during the 1981 plant outage. FRC is also aware of other test reports, referenced by other Licensees, that demonstrate satisfactory performance for a period of at least a few hours under inside-containment service conditions. FRC recommends that the Licensee contact the manufacturers to obtain access to these reports.

LICENSEE RESPONSE:

These valves are inside containment isolation valves for the emergency condenser, shutdown cooling, and cleanup systems. The valve actuators were supplied by Limitorque Corporation and are equipped with Reliance motors having Class B insulations. Our discussions with Limitorque personnel indicated that a test was performed by Franklin Institute Research Laboratories for Westinghouse Company utilizing the same valve assembly with motors having Class B insulation. According to the same source, the valve functioned at least 12 hours under conditions expected after a LOCA. The report was identified by the Limitorque personnel as FIRL test F-C2485-01 (dated May 1969). Several attempts by us to obtain this test report did not succeed since the report is classified as Westinghouse proprietary information. In view of this situation, a decision was made by JCP&L to replace all of these valves with qualified valve assemblies. Accordingly, purchase order No. 28930 was issued on December 20, 1979 and the valve assemblies, along with qualification report, were delivered to Oyster Creek Nuclear Generating Station in June 1980 and are currently kept in our storage room on site. Therefore, the qualified valve assemblies will be installed at the next scheduled shutdown, which will take place in the spring of 1981.

FRC EVALUATION:

Since the Licensee has not provided valid qualification documentation for this equipment, full qualification has not been established in accordance with the requirements of the Guidelines. Based upon a review of the originally cited reference for similar type valves and the equipment's brief required operating time, this equipment is expected to function adequately.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because, although valid qualification documentation has not been provided, test reports for similar type valves have shown that the equipment is likely to function adequately during an accident. The Licensee has committed to replace these valves by the spring of 1981.

4.5.2.3 Equipment Item No. 11
Temperature Detectors Located in the Reactor Building
Rochester Instrument System, Model Not Stated
Isolation Condenser Area Temperature Detectors (1B-06-E through H)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

The isolation condenser area temperature monitors provide indication in the control room of steam leaks in the area. These temperature detectors do not provide any automatic safety functions, but are referred to in the station emergency procedures as one of the parameters that can be used to detect leaks in the isolation condenser system. Since the system is primarily there to detect leaks and not breaks, it is unlikely that the area temperature will reach those levels described in the worst-case break analysis. The one-year worst-case integrated radiation exposure to these instruments is on the order of 1 Mrad for two detectors and on the order of 64×10^4 rads on the other two. While a material list is not available at this time, an evaluation of other temperature switches at the facility shows that radiation exposure up to 1 Mrad is acceptable. The evaluation uses a one-year exposure, and these instruments are used only to verify steam leaks in the area. They would only be utilized by the operators during the first few minutes of any event involving steam leaks in the main steam or isolation condenser system.

FRC EVALUATION:

FRC has reviewed the operational evaluation above and the information contained on SCEW sheets 44, 45, 46, and 47 and notes the following:

a. The Licensee states that the maximum temperature/pressure to which the equipment is exposed are 280°F/16 psia at the radiation levels noted above.

b. Note B of the SCEW sheets states that this equipment will either be qualified or replaced by July 1, 1982.

c. The equipment is required only for the first 10 minutes of a HELB.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because there is no evidence of qualification, but there is a high likelihood of operability based on the analysis provided by the Licensee. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to a program of equipment qualification or replacement by June 1982.

4.5.2.4 Equipment Item Nos. 19, 20, 21A, and 22A
Solenoid Valves Located in the Reactor Building
19: ASCO Model 8344-B27 (V-27-1, -2)
20: ASCO Model 8344-A27 (V-27-3, -4)
21A: ASCO Model 83148 (V-23-13)
22A: ASCO Model WP8300B61RU (V-23-17, -18)
Purge Valves and Nitrogen Valves
(Final Licensee References 2.7, 2.11 [Items 20, 21A, 22A], and 2.13 [Item 19])

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These are normally closed containment isolation valves that will not change position given a failure of the solenoid valve. They are in a non-harsh temperature/pressure environment. Our evaluation of the component materials revealed that this component contains thermal aging and radiation-sensitive materials (Buna-N and/or fish paper). Therefore, the sensitive component materials will be replaced by June 1982.

FRC EVALUATION:

References 2.11 and 2.13 provide information on the sensitivity of materials to nuclear radiations. Reference 2.7 is not adequately identified and a copy was not provided for review. As noted in Appendix D, this equipment should be qualified for a HELB environment. Also, FRC is not aware of valid qualification documentation for this solenoid valve from other sources. Therefore, qualification has not been established in accordance with the requirements of the Guidelines. It is expected that this equipment will function adequately because the environment is not extremely "harsh."

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report. The Licensee should proceed with the preventive maintenance activities on an expedited schedule. The manufacturer should be consulted to obtain recommended replacement schedules for the coils and other non-metallic components used in these valves.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b. Although valid qualification documentation has not been provided, the equipment is likely to function adequately because the environmental conditions are not harsh (except for radiation) for the accident it is intended to mitigate. The Licensee has stated that thermal- and radiation-sensitive materials will be replaced by June 1982.

4.5.2.5 Equipment Item No. 22B
Solenoid Valves Located in the Reactor Building
ASCO Model WP8300B61RU
Ventilation Valves (V-23-21, -22; V-28-17, -18, -47)
(Licensee References 2.7 and 2.11)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These are containment isolation valves that are normally closed and would not be required for outside-containment HELBs. Our evaluation of the component materials revealed that this component contains thermal aging and radiation-sensitive materials (Buna-N and/or fish paper). Therefore, the sensitive component materials will be replaced by June 1982.

FRC EVALUATION:

Reference 2.11 provides information on the sensitivity of materials to nuclear radiations. Reference 2.7 is not adequately identified and a copy was not provided for review. FRC is not aware of valid qualification documentation for this solenoid valve; therefore, qualification has not been established in accordance with the requirements of the Guidelines. It is expected that this equipment will function adequately because the only harsh environmental parameter is radiation. However, FRC notes that ASCO has provided recommended replacement schedules for the coils and elastomer parts used in these valves. The Licensee should ensure that the preventive maintenance program includes these recommended replacement schedules.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b. Although valid qualification documentation has not been provided, the equipment is likely to function adequately because the environmental conditions are not harsh (except for radiation) for the accident it is intended to mitigate. The Licensee has stated that thermal- and radiation-sensitive materials will be replaced by June 1982.

4.5.2.6 Equipment Item No. 26
Solenoid Valves Located in the Reactor Building
Atkomatic Model 15-702-B, Type 50R
Particulate Monitor System, Oxygen Analyzer System, and Torus Sample System Valves (V-38-16, V-38-17, V-38-9, and V-38-10)
(Final Licensee References 2.6 and 2.7)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These valves are in a non-harsh temperature/pressure environment and are not required to function for HELBs outside containment. Our evaluation of the component materials revealed that this component contains thermal aging and radiation-sensitive materials (Buna-N and/or fish paper). Therefore, the sensitive component materials will be replaced by June 1982.

FRC EVALUATION:

The references cited by the Licensee are not adequately identified and copies were not provided for review. Also, FRC is not aware of valid qualification documentation for this solenoid valve from other sources. Therefore, qualification has not been established in accordance with the requirements of the Guidelines. However, this equipment is likely to function adequately because the only harsh environmental parameter is radiation.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b. Although valid qualification documentation has not been provided, the equipment is likely to function adequately because the environmental conditions are not harsh (except for radiation) for the accident it is intended to mitigate. The Licensee has stated that thermal- and radiation-sensitive materials will be replaced by June 1982.

4.5.2.7 Equipment Item No. 27
Solenoid Valves Located in the Reactor Building
ASCO Model LB82627
Particulate Monitor System, Oxygen Analyzer System, and Torus Sample System Valves (V-38-22 and V-38-23)
(Final Licensee Reference 2.7)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These valves are in a non-harsh temperature/pressure environment and are not required to function for HELBs outside containment. Our evaluation of the component materials revealed that this component contains thermal aging and radiation-sensitive materials (Buna-N and/or fish paper). Therefore, the sensitive component materials will be replaced by June 1982.

FRC EVALUATION:

The reference cited by the Licensee is not adequately identified and copies were not provided for review. Also, FRC is not aware of valid qualification documentation for this solenoid valve from other sources. Therefore, qualification has not been established in accordance with the requirements of the Guidelines. This equipment is likely to function adequately because the only harsh environmental parameter is radiation. However, FRC notes that ASCO has provided recommended replacement schedules for coils and elastomer parts used in these valves. The Licensee should ensure that the preventive maintenance program includes these recommended replacement schedules.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b. Although valid qualification documentation has not been provided, the Licensee has shown that the equipment is likely to function adequately because the environmental conditions are not harsh except for radiation for the accident it is intended to mitigate. The Licensee has stated that the thermal- and radiation-sensitive materials will be replaced by June 1982.

4.5.2.8 Equipment Item No. 28
Temperature Switches Located in the Steam Tunnel
Fenwal Model 17002-40
Reactor Isolation Temperature Switches for Main Steam Line Leak Detection (IB-10 A through P)
(Final Licensee Reference 2.11)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These temperature switches are located in the main steam line tunnel outside the drywell to detect a MSLB in the tunnel. However, the detections of a MSLB are provided by other redundant and diverse signals that are not affected by the break. Those are reactor water low-level signals, main steam line low-pressure signals, and main steam line high-flow signals.

- o Qualification documentation is not available at this time.
- o This equipment will be either replaced or qualified by June 1, 1982.

FRC EVALUATION:

The Licensee has neither submitted nor referenced qualification documentation for this item.

The Licensee has stated:

- o The switches are redundant to other safety-related equipment which is not simultaneously exposed to the MSLB harsh environment.
- o This equipment is required to operate during a HELB outside containment.
- o This equipment will be qualified or replaced by June 1, 1982.

FRC notes that the components are differential expansion thermoswitches, non-indicating and hermetically sealed, with adjustable setpoint and NEMA and housing which provides a high temperature trip signal to the reactor protection system.

FRC has reviewed documentation relevant to this equipment item for the environmental qualification review program and has reached the following conclusions:

- A Fenwal model switch was tested.
- was performed at dry heat load. The setpoint retained a .
- A submergence test was conducted at psig.
- A radiation test imposed a dose of Mrd.
- A high temperature test (heated aluminum block) subjected the switch to for .

FRC concludes that the heated aluminum block (dry heat) and immersion tests were not equivalent to HELB high temperature all-steam testing. However, the radiation test imposed a greater dose than the required 6.1×10^4 rd. It should also be noted that the test specimen model was No. 17023-6, whereas the actual installed equipment model is 17002-40.

FRC concludes that this component lacks documentation of operability under HELB environmental service conditions. A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b. Although the qualification documentation is deficient with respect to HELB (high temperature all-steam) testing and the specific relationship of the installed switches to the test specimen, the equipment is highly likely to operate. Its design is simple, the adverse environment is within the temperature range in which the unit has performed satisfactorily, and the immersion test provides assurance that steam in-leakage will not be a problem. However, aging and qualified life have not been addressed. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to equipment qualification or replacement by June 1982.

4.5.2.9 Equipment Item Nos. 31B (previously designated I-A1) and 32B (previously designated I-B1)
Solenoid Valves Located Within the Drywell
31B: ASCO Model 206-832-3RU
32B: ASCO Model 206-301-3RU
Main Steam Isolation Valves
(Original Licensee References 2.2, 2.3, and 2.24;
Final Licensee References 2.16 and 2.24)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.2.1):

Reference 2.24 is a proprietary test report describing a qualification program conducted for a number of ASCO solenoid valves. DITER References 2.2 and 2.3 are letters from ASCO documenting that the tested and installed equipment models have the same coils, coil enclosures, and valve seats. FRC comments as follows, based on review of these references:

a. Of the valve models tested, those with model numbers that correspond to those of the installed equipment are:

   o Items I-A1 and I-B1: Sample No. 4, Model No. having a

   o Item I-C1: Sample No. 5, Model No. having a ,

   The three references establish conformance between the tested and installed equipment.

b. The environmental and operational service condition parameters used in the qualification test program exceeded those dictated by plant-specific requirements in all cases except (i) the of the steam temperature/pressure profile and (ii) the use of a boric acid/sodium hydroxide spray solution in lieu of a sodium dichromate solution. These deficiencies are not significant. The Licensee submittal did not explicitly consider the nuclear radiation dose resulting form beta radiations (including the bremsstrahlung radiation it creates while being attenuated). Because the nonmetallic components of the solenoid valves are encased within metallic enclosures, the dose contribution from beta radiation can be expected to be quite small. The test program included a sufficiently large gamma radiation dose ( ) that the beta dose contribution can be considered to have been accommodated.

c. The pre-aging simulated in the test program was intended to represent an installed life (and hence a qualified life) of ambient temperature. Reference 2.24 states that the coil and seats should be replaced at intervals. Provided that the Licensee has established (i) a replacement schedule consistent with this requirement and (ii) a program to review any in-service failures to determine whether they are caused by aging degradation, the equipment is considered to be qualified with a qualified life of 4 years.

LICENSEE RESPONSE:

[No response provided.]

FRC EVALUATION:

FRC has reviewed the reference(s) cited by the Licensee and has the following comments:

1. During the qualification test program described in Reference 2.24,

   The results of the test must therefore be regarded as inconclusive until the uncertainties associated with the method of making the wiring interface with the solenoid, both in the plant and in the test, are resolved. The Guidelines state (Section 5.2.5):

   > "If a component fails at any time during the test, even in a so called 'fail safe' mode, the test should be considered inconclusive with regard to demonstrating the ability of the component to function for the entire period prior to the failure."

   They further state (Section 5.2.6):

   > "The equipment mounting and electrical or mechanical seals used during the type test should be representative of the actual installation for the test to be considered conclusive."

2. The pre-aging simulated in the test program was intended to represent an installed life (and hence a qualified life) of ambient temperature. The ambient temperatures at the installed locations within the plant are lower, and hence the qualified life is longer. An explicit, conservative determination of qualified life and a replacement schedule (if needed) must be established.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b. Although the results of the qualification test program are inconclusive, the required function occurs early in the accident scenario, and is therefore highly likely to be performed properly. The Licensee should determine how the electrical connections are sealed, establish that moisture infiltration will not cause failure, and establish a conservative qualified life. A surveillance program should be implemented to monitor performance and identify any degradation which would indicate the need for maintenance or replacement.

4.5.2.10 Equipment Item Nos. 34C (previously designated I-2B) and 44 (previously designated I-2D)
Motorized Valve Actuators Located Within the Drywell
34C: Limitorque Model SMB-0 with Reliance Motor (Class B Insulation) Shutdown Cooling Valves (V-17-19 and V-16-1)
44: Limitorque Model SMB-2 with Reliance Motor (Class B Insulation) Isolation Condenser Valves (V-14-36, -37)
(Licensee Reference 2.4)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.1):

Reference 2.4 is a test report of a qualification test for an SMB-0 actuator. FRC has the following comments:

a. The test report is for a Limitorque Model SMB-0 actuator with a Reliance motor having Class B insulation. The Guidelines require that the test specimens be the same as the equipment being qualified. The Licensee did not present an analysis comparing the impact of deviations between the test specimen's specific design features, materials, and production procedures and those of the installed equipment. Therefore, an independent conclusion cannot be reached regarding the extent to which the tested equipment is similar to that installed in the plant, and the validity of the test, as evidence of qualification, has not been established.

b. The test program included wear/thermal/humidity/seismic aging, vibration tests to simulate severe seismic events, and a steam exposure. The environmental parameters, aging considerations, and other aspects of the test program were intended to demonstrate qualification of equipment located outside of the primary containment; they are not adequate to demonstrate qualification for these equipment items located within the containment drywell.

The Licensee has stated that these equipment items will be replaced with fully qualified equipment during the 1981 plant outage. FRC is also aware of other test reports, referenced by other Licensees, that demonstrate satisfactory performance for a period of at least a few hours under inside containment service conditions. FRC recommends that the Licensee contact the manufacturers to obtain access to these reports.

LICENSEE RESPONSE:

> These valves are inside containment isolation valves for emergency condenser, shutdown cooling, and cleanup systems. The valve actuators were supplied by Limitorque Corporation and are equipped with Reliance motors having Class B insulations. Our discussions with Limitorque personnel indicated that a test was performed by Franklin Institute Research Laboratories for Westinghouse Company utilizing the same valves assembly with motors having Class B insulation. According to the same source, the valve functioned at least 12 hours under conditions expected after a LOCA. The report was identified by the Limitorque personnel as FIRL test F-C2485-01 (dated May 1969). Several attempts by us to obtain this test report did not succeed since the report is classified as Westinghouse proprietary information. In view of this situation, a decision was made by JCP&L to replace all of these valves with qualified valve assemblies. Accordingly, purchase order No. 28930 was issued on December 20, 1979 and the valve assemblies, along with qualification report, were delivered to Oyster Creek Nuclear Generating Station in June 1980 and are currently kept in our storage room on site. Therefore, the qualified valve assemblies will be installed at the next scheduled shutdown, which will take place in the spring of 1981.

FRC EVALUATION:

The Licensee has not provided valid qualification documentation for this equipment. Therefore, full qualification has not been established in accordance with the requirements of the Guidelines. Based upon a review of the originally cited reference and the brief time this equipment must operate, the equipment is expected to function adequately.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because, although valid qualification documentation has not been provided, the Licensee has shown that the equipment is likely to function adequately. The Licensee has committed to replace this equipment with fully qualified equipment in the spring of 1981.

4.5.2.11 Equipment Item Nos. 35 and 36
Solenoid Valves Located in the Reactor Building
35: ASCO Model LM831424
36: ASCO Model WP8300B61V
Drywell Isolation Valves
(Final Licensee References 2.7 and 2.11)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

35: Reactor Water Sample Valves (V-24-30)

This valve is the outside containment isolation valve for the reactor sample line. Although this valve may see a fairly high temperature environment in the event of a cleanup line break, it is normally closed. In addition the redundant valve inside containment is also normally closed. In the event it was open, both the inside and outside containment valve would be closed on diverse containment isolation signals. Our evaluation of the component materials revealed that this component contains thermal aging and radiation-sensitive materials (Buna-N and/or fish paper). Therefore, the sensitive component materials will be replaced by June 1982.

Based on the above consideration, it is unlikely that containment isolation would not be achieved via the sample line for a cleanup line break.

36: Drywell Sump Discharge Valves (V-22-1, V-22-2, V-22-28, and V-22-29)

These valves are the containment isolation valves for the Drywell equipment drain tank and sump. These valves do not see a harsh temperature/pressure environment for any postulated HELBs. Also, it should be noted that these valves are not needed for isolation purposes for breaks outside containment. Our evaluation of the component materials revealed that this component contains thermal aging and radiation-sensitive materials (Buna-N and/or fish paper). Therefore, the sensitive component materials will be replaced by June 1982.

Based on the above information, the isolation function of these valves is maintained for all postulated HELBs outside containment.

FRC EVALUATION:

The references cited by the Licensee are not adequately identified and copies were not provided for review. Also, FRC is not aware of valid qualification documentation for this solenoid valve from other sources. Therefore, qualification has not been established in accordance with the requirements of the Guidelines.

FRC has reviewed the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment (see Appendix D of this report) and is satisfied with the technical discussion except for a remaining concern about the need to open the valves after the accident has occurred. FRC also notes that ASCO has provided recommended replacement schedules for coils and elastomeric components used in these valves. The Licensee should ensure that the preventive maintenance program includes these recommended replacement schedules.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because, although valid qualification documentation has not been provided, the Licensee has shown that the equipment is likely to function adequately during the early stages of an accident. Maintenance and replacement of parts or the entire unit in accordance with the manufacturer's schedule should be followed. The Licensee has stated that thermal- and radiation-sensitive materials will be replaced by June 1982.

4.5.2.12 Equipment Item No. 37
Motorized Valve Actuators Located in the Reactor Building
Limitorque Model SMB-1 with Reliance Motor (Class B Insulation)
Core Spray Valves (V-20-15, -21, -40, -41)
(Final Licensee References 2.2, 2.3, 2.4, and 2.5)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

The core spray system is set up such that V-20-15 and V-20-40 are located in parallel on one side of the reactor, and V-20-21 and V-20-41 are located on the other side in parallel approximately 180 degrees apart and on two different floors. Also, only one of the valves needs to operate for the system to perform its function. Only one pair of valves will be subjected to the harsh accident conditions, thereby leaving the other pair in a relatively mild environment environment and able to function.

[The Licensee also notes that qualification of these units is not established by References 2.3 and 2.4.]

FRC EVALUATION:

The Licensee's references have been discussed in connection with Equipment Item Nos. 3A, 3B, 4A, 4B, 34A, and 34B. As the Licensee notes, the cited test reports do not establish qualification in accordance with the requirements of the Guidelines. Based upon a review of the Licensee Response and all known test reports that may apply to this equipment, it appears likely that this equipment will function adequately.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because, although valid qualification documentation has not been provided, the extensive amount of testing conducted on similar equipment provides reasonable assurance that the equipment is likely to function adequately. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to a program of equipment qualification or replacement by June 1982.

4.5.2.13 Equipment Item No. 40
Motorized Valve Actuators Located in the Reactor Building
Limitorque Model SMB-2 with Reliance and Peerless Motors (Class B Insulation)
Emergency Condenser Valves (V-14-30 through -35)
(Final Licensee References 2.2, 2.3, 2.4, and 2.5)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

The isolation condenser system is set up such that V-14-30 and V-14-31 are connected in series; V-14-32 and V-14-33 are also connected in series. According to calculations performed during NEMA standards (Pub. No. mg/1) on motor isolation, V-14-30 and V-14-32, being ac class B motors, can withstand a maximum ambient temperature of 221°F; whereas the others (V-14-31, V-14-33, and V-14-34, V-14-35), being dc Class B motors, can withstand a maximum ambient temperature of 275°F. According to our analysis, the maximum accident temperature is 280°F. The above mentioned motors (V-14's) are only needed for a maximum of 60 seconds. Therefore, the motor will have performed its function 60 seconds into the accident and will no longer be needed. It is unlikely that the motor windings will heat up (due to the accident temperature) to this critical temperature of 275° within the time that the motors are needed.

[The Licensee also notes that qualification of these units is not established by Reference 2.3 and 2.4.]

FRC EVALUATION:

The Licensee's references have been discussed in connection with Equipment Items 3A, 3B, 4A, 4B, 34A, and 34B. As the Licensee notes, the cited test reports do not establish qualification in accordance with the requirements of the Guidelines. Based upon a review of the Licensee Response and all known test reports that may apply to this equipment, it appears likely that this equipment will function adequately.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because, although valid qualification documentation has not been provided, the extensive amount of testing conducted on similar equipment provides reasonable assurance that the equipment is likely to function adequately during the brief required operating time. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to a program of equipment qualification or replacement by June 1982.

4.5.2.14 Equipment Item No. 42
Solenoid Valve Located in the Reactor Building
ASCO Model WT8300B61RV
Head Cooling System Isolation Valve (V-31-2)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

The purpose of this valve is to provide reactor coolant boundary isolation. This valve is normally closed, and fails closed on a loss of air or power. Also, the piping outside of the containment is designed for a higher pressure than the Nuclear Steam Supply System. This valve is used if the head cooling system needed to ensure the Technical Specification limit on the vessel flange to head temperature of 200°F was not violated during a plant cooldown.

Based on the above considerations, it is expected that the valve will continue to carry out its safety function of isolating a reactor coolant system boundary even in the event of a HELB inside or outside containment. Our evaluation of the component materials revealed that this component contains thermal aging and radiation-sensitive materials (Buna-N and/or fish paper). Therefore, the sensitive component materials will be replaced by June 1982.

FRC EVALUATION:

The Licensee has not provided, and FRC has found no other sources of, valid qualification documentation for this solenoid valve. Therefore, qualification has not been established in accordance with the requirements of the Guidelines.

FRC's review of the Licensee's justification for (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report. It is expected that this equipment will function adequately because the only harsh environmental condition is radiation. However, FRC notes that ASCO has provided recommended replacement schedules for coils and elastomer components used in these valves. The Licensee should ensure that the preventive maintenance program includes the recommended replacement schedules.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because, although valid qualification documentation has not been provided, the equipment is likely to function adequately. The Licensee has stated that thermal- and radiation-sensitive materials will be replaced by June 1982.

4.5.2.15 Equipment Item No. 43 (previously designated I-1C)
Solenoid Valve Located Within the Drywell
ASCO Model NP-8320A187E
Sample Valve
(Original Licensee References 2.2, 2.3, and 2.24;
Final References 2.16 and 2.24)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.2.1):

Reference 2.24 is a proprietary test report describing a qualification program conducted for a number of ASCO solenoid valves. DITER References 2.2 and 2.3 are letters from ASCO documenting that the tested and installed equipment models have the same coils, coil enclosures, and valve seats. FRC comments as follows, based on review of these references:

a. Of the valve models tested, those with model numbers that correspond to those of the installed equipment are:

   o for Items IA-1 and IB-1: Sample No. 4, Model No.

   o Item IC-1: Sample No. 5, Model No.

   The three references establish conformance between the tested and installed equipment.

b. The environmental and operational service condition parameters used in the qualification test program exceeded those dictated by plant-specific requirements in all cases except (i) the of the steam temperature/pressure profile and (ii) the use of a boric acid/sodium hydroxide spray solution in lieu of a sodium dichromate solution. These deficiencies are not regarded as significant. The Licensee submittal did not explicitly consider the nuclear radiation dose resulting from beta radiations (including the bremsstrahlung radiation it creates while being attenuated). Because the nonmetallic components of the solenoid valves are encased within metallic enclosures, the dose contribution from beta radiation can be expected to be quite small. The test program included a sufficiently large gamma radiation dose ( ) that the beta dose contribution can be considered to have been accommodated.

c. The pre-aging simulated in the test program was intended to represent an installed life (and hence a qualified life) of ambient temperature. Reference 2.24 states that the coil and seats should be replaced at intervals. Provided that the Licensee has established (i) a replacement schedule consistent with this requirement and (ii) a program to review any in-service failures to determine whether they are caused by aging degradation, the equipment is considered to be qualified with a qualified life of 4 years.

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

[No response provided.]

FRC EVALUATION:

FRC has reviewed the references cited by the Licensee and has the following comments:

1. During the qualification test program described in the reference,

The results of the test must therefore be regarded as inconclusive until the uncertainties associated with the method of making the wiring interface with the solenoid, both in the plant and in the test, are resolved. The Guidelines state (Section 5.2.5):

"The equipment mounting and electrical or mechanical seals used during the type test should be representative of the actual installation for the test to be considered conclusive."

2. The pre-aging simulated in the test program was intended to represent an installed life (and hence a qualified life) of ambient temperature. The ambient temperatures at the installed locations within the plant are lower, and hence the qualified life is longer (see Section 4.1.3).

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because, although valid qualification documentation has not been provided, the equipment is expected to function to close and remain closed during the early portion of an accident. To fully quality this equipment, the Licensee should demonstrate that the electrical connection is adequately sealed, and should also demonstrate long-term performance. The qualified life should be determined on a more conservative basis.

4.5.2.16 Equipment Item Nos. 46 and 47 (previously designated I-4A, B, C, D)
Electrical Connectors Located Within the Drywell Containment
ITT-Cannon Models
46: CA-3106E-36A-46P-F80
CA-3100K-36A-46S-F80
47: CA-06RX-36A-10P-A95
CA-3100RX-36A-10S-A95
(Final Licensee References 2.7 and 2.16)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.3):

Licensee Reference 2.7 is a report on a qualification test performed on connectors that are virtually identical to those installed in the plant. FRC's comments on this test report are as follows:

a. A thorough analysis was made (and reported in Reference 2.7) of the operation service conditions associated with the installed equipment and of thermal aging effects. The Licensee's contractor concluded that the connectors are not subject to aging degradation, but the basis for this claim is not rigorous (i.e., it relied on the claim of 40,000 hours service at 105°F and the "10°C Rule," rather than on specific aging data). Possible long-term effects of humidity and nuclear radiation were not considered. A more conservative approach to qualified life should be taken.

b. The temperature/pressure profile in the test exceeded the plant-specific profile (except for rise time), and the correct chemical spray was used.

c. The analysis in the report concluded that only 4.8 Mrd of nuclear radiation would be required to establish qualification. This is regarded as inadequate for equipment that must provide long-term service within the drywell. Also, it is not stated in the report that even this rather modest exposure was administered.

LICENSEE RESPONSE:

[No response provided.]

FRC EVALUATION:

The Licensee provided no response to the DITER; therefore, the original comments apply. The Licensee states in the SCEW sheets that this equipment will be qualified or replaced by July 1, 1982. The major qualification discrepancy is between the 57-Mrd radiation dose required by the SCEW sheet and the test dose of less than 5 Mrd.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because the equipment is highly likely to perform adequately, but the qualification is not complete for the radiation dose specified by the Licensee. It is noted that the Licensee has committed to qualify or replace the item by June 1982.

4.5.2.17 Equipment Item No. 51 (previously designated I8)
Electrical Cable Located Within the Drywell
Tensolite, Model Not Stated (previously stated "Tefzel Insulation/Unjacketed")
(Final Licensee References 2.16 and 2.22)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.4.1):

Documentation reflecting qualification for the following equipment has not been made available for review.

LICENSEE RESPONSE:

[No response provided.]

FRC EVALUATION:

Licensee Reference 2.22 and other test reports available to FRC on Tefzel insulated cables (FIRL Report F-C3859-1) have been reviewed. Based on these reviews, FRC has the following comments:

a. With regard to similarity of test specimen to installed cable, the Licensee submittal has not identified the type, size, or number of conductors, or the jacket material of the Tensolite cable.

b. Reference 2.22 notes that 7C AWG No. 12 with a combined and Nomex insulation and Tefzel jacket were tested to recommendations of IEEE Stds 323-74 and 383-74 and had satisfactory insulation resistance after test. No data were presented on insulation resistance during LOCA exposure nor does the reference state that the tested cables are the same as those installed at Oyster Creek (see Appendix G).

c. Single conductor and multiconductor cables using Tefzel insulation for another manufacturer did not perform satisfactorily as reported in FIRL Report and did not survive the test. It appears that this was due to aging and a 200-Mrd radiation exposure.

d. Pressure, temperature, humidity, and chemical spray of the tests enveloped the LOCA conditions for Oyster Creek.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because extensive testing shows it is highly likely to operate satisfactorily at Oyster Creek where the maximum exposure is 57 Mrd. It is recommended that this equipment be replaced with cable that fully satisfies the DOR Guidelines.

4.6 NRC Category V
EQUIPMENT THAT IS UNQUALIFIED

The DOR Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology and program be referenced and made available for review of all Class 1E equipment.

The qualification of equipment items in this section has been judged to be deficient or inadequate, based upon review of the documentation provided by the Licensee. The extent to which the equipment items fail to satisfy the criteria of the DOR Guidelines can be categorized as follows: (1) documentation reflecting qualification as specified in the DOR Guidelines has not been made available for review, (2) the documentation is inadequate, or (3) the documentation indicates that the equipment item has not successfully passed required tests.

4.6.1 Equipment Item No. 6
Pressure Transmitters Located in the Reactor Building
General Electric Model GE/MAC 551
Reactor Vessel Pressure Transmitter (ID-45A and B)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

> These pressure transmitters are installed to provide only an indication to the Control Room operator. The transmitters do not perform any safety functions. Even if these transmitters failed, the relief valves in ADS or 16 safety valves will relieve the pressure in the vessel, and thus the reactor vessel is well protected from over pressurization for any postulated HELB.

FRC EVALUATION:

The Licensee has neither submitted nor referenced qualification documentation for this item. Also, FRC is not aware of qualification documentation for this equipment from other sources. Therefore, qualification has not been established in accordance with the requirements of the Guidelines.

The Licensee Response states that the transmitters do not perform a safety function. However, these transmitters do provide the operator an indication of reactor vessel pressure. Since this information is necessary for cold shutdown and to allow the operator to monitor the performance of safety systems (i.e., the automatic depressurization system [ADS], the transmitters are safety-related.

FRC concludes that this equipment lacks documentation demonstrating operability under HELB environmental service conditions. The Licensee has provided justification for interim operation by stating that relief valves in the ADS will relieve vessel pressure and protect against over-pressurization (see Appendix D of this report). The Licensee also stated that this equipment will be replaced or qualified by June 1, 1982.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because there is no evidence of qualification. FRC's review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to equipment qualification or replacement by June 1982.

4.6.2 Equipment Item No. 7
Pressure Transmitter Located in the Reactor Building
General Electric Model VPF 1438
Reactor Vessel Pressure Transmitter (1D-46 A and B)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These pressure transmitters are installed to provide only an indication to the Control Room operator. The transmitters do not perform any safety functions. Even if these transmitters failed, the relief valves in ADS or 16 safety valves will relieve the pressure in the vessel, and thus the reactor vessel is well protected from over pressurization for any postulated HELB.

FRC EVALUATION:

The Licensee has neither submitted nor referenced qualification documentation for this item. Also, FRC is not aware of qualification documentation for this equipment from other sources. Therefore, qualification has not been established in accordance with the requirements of the DOR Guidelines.

The Licensee response states that the transmitters do not perform a safety function. However, these transmitters provide the operator with an indication of reactor vessel pressure. Since this information is necessary for cold shutdown and to allow the operator to monitor the performance of safety systems (i.e., the ADS system), the transmitters are safety-related.

FRC concludes that this component lacks documentation demonstrating operability under HELB environmental service conditions. The Licensee has provided justification for interim operation by stating that relief valves in ADS will relieve vessel pressure and protect against over-pressurization (see Appendix D of this report). The Licensee also stated that this equipment will be replaced or qualified by June 1, 1982.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because there is no evidence of qualification. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to equipment qualification or replacement by June 1982.

4.6.3 Equipment Item Nos. 8A and 8C
Transmitters Located in the Reactor Building
General Electric Model GE/MAC 553
8A: Emergency Condenser Level (1G-06-A-1, -A-2, -B-1, -B-2)
8C: Containment Spray Flow (IP-03A, B)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

Emergency Condenser Level Transmitter:

Each emergency condenser, containing a minimum water volume of 22,730 gallons of condensate on shell side, provides 11,060 gallons above the top of the tube handles. This volume can accommodate the reactor decay heat for up to 1 hour and 40 minutes without any need for makeup water (both condensate and service). If one condenser is used, it can accommodate reactor decay heat up to 45 minutes after a scram from full power before makeup is required. The reactor can also be depressurized by using ADS. Therefore, the operator can manually initiate the ADS actuation with 45 minutes of a scram following an accident. The ADS, which is located inside the Drywell, is not affected by the accident, since the worst-case HELB considered is an emergency condenser line break outside the Drywell.

Containment Spray Flow Transmitter:

The containment spray flow transmitters are used by the control room operator to verify containment spray system is actually delivering its required flow. The containment spray system would only be used if there had been an inside containment LOCA or the torus had to be utilized as a heat sink in order to achieve safe shutdown. In the case of an inside containment LOCA the harsh temperature and pressure environment outside containment would not exist and only radiation effects would have to be considered. For HELB's outside containment only IP-03-B would see a slightly harsh temperature of 140 degrees. There is not documentation of radiation qualification for these components. It should be noted that these instruments provide only indication and do not perform any automatic safety functions. Even considering the loss of this indication the operator has various other backup parameters that will verify adequate system flow. They are containment spray motor amperes, pump discharge pressure, torus temperature and valve position.

Based upon the above justification, it is expected that instruments will function as intended if they were required for core spray system flow verification.

FRC EVALUATION:

The Licensee has neither submitted nor referenced qualification documentation for these items. Also, FRC is not aware of qualification documentation for this equipment from other sources. Therefore, qualification has not been established in accordance with the requirements of the Guidelines.

FRC concludes that this component lacks documentation demonstrating operability under HELB environmental service conditions. The Licensee has provided a justification for interim operation by stating that (1) the emergency condenser minimum volume of water can accommodate reactor decay heat for a short time and the operator can actuate the ADS, and (2) backup instrumentation can indicate spray flow (see Appendix D of this report).

The Licensee states that this equipment will be replaced or qualified by June 1, 1982.

FRC CONCLUSION:

These equipment items are assigned to NRC Category V because there is no evidence of qualification. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to equipment qualification or replacement by June 1982.

4.6.4 Equipment Item Nos. 8B, 8D, and 8E
Transmitters Located in the Reactor Building
General Electric Model GE/MAC 553
8B: Reactor Water Level Transmitters (1D-13A, B; 1A-12A, B)
8D: Drywell Pressure Transmitter (IP-07)
8E: Containment Spray Differential Pressure Transmitter (IP-05A through D)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

8B: Reactor Water Level Transmitters (1D-13A, B; IA-12A, B)
[SCEWS Nos.: 30-33]

These level transmitters are installed to provide only an indication to the control room operator and they do not perform any safety functions. As described in our justification for item 25, the reactor will be scramed and isolated regardless of the availability of these transmitters.

8D: Drywell Pressure Transmitter (IP-07) [SCEWS No.: 54]

The peak temperature and pressure seen by the switch are 230°F and 16 psia, respectively, following an Emergency Condenser line break outside containment (worst case HELB). The containment (Drywell) pressure transmitter is provided to transmit containment pressure indication to Control Room. However, it is not required to mitigate a line break outside containment.

8E: Containment Spray Differential Pressure Transmitter (IP-05A through D) [SCEWS Nos.: 133-136]

The purpose of these differential pressure transmitters is to detect tube leaks in the containment spray heat exchangers. These leaks might provide a potential leakage path to the environment of radioactive effluent. This component does not provide any automatic function and only serves to provide an alarm in the control room. It is not expected that the containment spray heat exchanger tubes would leak, since they were retubed with titanium in the spring of 1980. This material has proved to be highly resistant to corrosion in other similar applications at Oyster Creek.

Therefore, based on the above discussion, there is reasonable assurance that the containment spray heat exchanger will not provide an undetected leakage path for radioactive effluent.

[With respect to Equipment Items 8D and 8E, the Licensee has grouped this equipment with items for which the following statement made in the introductory paragraphs of Chapter 7, Reference 1, apply.]

These components are not required to mitigate a HELB. Even if this equipment were to fail after a HELB, the protection of the reactor is adequately provided by other systems (and the non-asterisked equipment). Therefore, we have evaluated the thermal aging and radiation susceptibility characteristics of the component materials. This evaluation revealed that certain equipment included thermal aging and radiation-sensitive materials (Buna-N and fish paper). JCP&L will replace this component material with a qualified one by June 1982.

FRC EVALUATION:

The adverse environmental conditions stated by the Licensee for these components are:

8B: 230°F/16 psia, 6.1×10^4 rd

8D: 230°F/16 psia, 3.9×10^5 rd

8E: 77-140°F/15 psia, 7.5×10^5 rd

FRC concludes that this equipment is exposed to a harsh environment and therefore must be qualified.

FRC notes that the Licensee SCEW sheets state that the equipment will be replaced or qualified by July 1, 1982, but the introduction to Chapter 7 [1] states that radiation-sensitive material will be replaced by July 1, 1982.

The Licensee has neither submitted nor referenced qualification documentation for this equipment. Also, FRC is not aware of qualification documentation for this equipment from other sources. Therefore, qualification has not been established in accordance with the requirements of the Guidelines.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

These equipment items are assigned to NRC Category V because there is no evidence of qualification. The Licensee has stated that the equipment item will be qualified or replaced with qualified equipment or that radiation- or thermal-sensitive material will be replaced by July 1, 1982.

4.6.5 Equipment Item No. 30
Electric Motors Located in the Reactor Building
General Electric Model 5K-818842A103
Drives Containment Spray Pumps (PM-51-1-1 through -4)
(Final Licensee Reference 2.14)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

"Qualified"

FRC EVALUATION:

The Licensee has referenced a General Electric test report and has stated that it will be made available for review, but has not yet done so.

The containment spray pump motors are located in a harsh area subject to a 165°F temperature and radiation dose of 1 Mrd. The Licensee's qualification documentation should verify that the motors' bearings, lubrication, insulation system, and motor lead splices will not be degraded by the harsh environment.

The Licensee should obtain and analyze maintenance information to determine if equipment degradation has been abnormal and to help estimate the equipment's qualified life (see Section 4.13).

FRC CONCLUSION:

The containment spray pump motors are assigned to NRC Category V because qualification documentation has not been provided. The Licensee should provide the referenced General Electric test report, and establish a conservative qualified life.

4.6.6 Equipment Item No. 45 (previously designated I-3A, -B, -C)
Electrical Penetration Located Within the Drywell
General Electric Models F01, NS02, NS03, and NS04
(Final Licensee References 2.16, 2.17, 2.18, and 2.19)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.2):

In general, electrical penetrations perform two safety-related functions: (i) provide a leak-tight barrier as part of the overall plant primary containment, minimizing release of radioactive materials, and (ii) carry electric power, control, and instrumentation signals across the containment boundary. With regard to the first function, the design of this equipment item has three implicit failure modes that must be addressed: distortion of the penetration structural members, failure of elastomeric seals on the mounting flange (if present), and failure of the seals and electrical insulation around individual conductors. With regard to the second function, two failure modes are relevant: breakdown of the electrical insulation, causing a short circuit to ground or between conductors (or high leakage currents, in the case of conductors for instrumentation signals), and breakage of the conductor, causing an open circuit. It is important to note that the two functions are related in at least two ways. First, two of the failure modes for the first function are likely to also cause one or both of the possible failure modes associated with the second function (i.e., an insulation or seal failure around a conductor may both impair containment integrity and cause electrical failures). Second, the fact that the conductors carry electrical currents results in higher than ambient temperatures in the seals and insulation and in electromagnetic and thermal-induced forces being imposed on these materials and the conductors. These effects help to induce failure modes, leading to impairment of both basic functions.

The environmental service conditions inside containment are more severe than those outside, considering both normal operation and possible accidents. Hence, these constitute the conditions for which qualification must be established.

FRC has reviewed the documentation submitted by the Licensee and has found it to be highly fragmented and deficient in several aspects, as follows:

a. While the Licensee claims that the penetrations supplied and installed in Oyster Creek are type tested in DITER Reference 2.6, no supporting documentation has been provided. Also, FRC notes that there is no identification of the type or models tested in this reference. There is likewise no traceability to the unit tested in Reference 2.17. The Guidelines require that the test specimen must be the same as the equipment being qualified. The Licensee did not present an analysis comparing the impact of deviations between the test specimen's specific design features, materials, and production procedures and those of the installed equipment. Therefore, an independent conclusion cannot be reached regarding the validity of the tests described in the referenced documentation.

b. For materials subject to thermal aging, the Guidelines require that qualified life must be established. The used in the penetrations is not identified in DITER Reference 2.6 and, while humidity aging tests were performed on several epoxies, no thermal aging test was reported on either the used or the penetration as a whole. The Guidelines require that thermal aging (where applicable), radiation exposure, chemical spray, LOCA/HELB testing, and submergence testing (where applicable) be conducted on the same sample(s).

c. Although the temperature/pressure profile used in the test reported in Reference 2.17 exceeded the plant-specific profile, no spray was used and there were no nuclear radiation tests on ; it has not been established that this material is used in the units installed in the plant.

d. The tests reported in Reference 2.17 included some electric current loadings, but no information has been presented to show that the values used are adequate for the plant, especially considering possible short circuits in high power conductors as the "active single failure."

LICENSEE RESPONSE:

[No response provided.]

FRC EVALUATION:

The Licensee has identified the electrical penetrations as Types F01 and/or NS02, NS03, NS04. Since the Licensee has provided no additional information, the comments in the DITER still apply. The Licensee notes that the penetrations will be either qualified or replaced by July 1, 1982.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because testing has not demonstrated that the equipment will meet Guidelines requirements. The Licensee has committed to qualify or replace the equipment by July 1982.

4.6.7 Equipment Item No. 48 (previously designated I5)
Terminal Blocks Located Within the Drywell
General Electric Model EB
(Final Licensee References 2.16 and 2.20)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.4):

Reference 2.20 is a brief report that describes the results of a steam exposure test on an exposed General Electric CR 151B terminal block, plus one from another manufacturer. DITER Reference 2.10 is a General Electric Co. letter stating that there is very little difference between the CR 151B and EB terminal blocks. This letter also claims that the materials have good tolerance to nuclear radiations, but provides no evidence to substantiate either of these claims. DITER Reference 2.17 is a report on steam exposure tests conducted on General Electric EB-25 terminal blocks. FRC's review of these qualification documents has resulted in the following findings:

a. The Guidelines require that the model of the tested unit be the same as that of the equipment being qualified. The type test is valid only if the installed equipment and tested unit have the same design and materials and closely similar production procedures and stress levels. The Licensee has neither completely identified the installed equipment nor presented an analysis comparing the impact of deviations between the test specimen's design features, materials, and production procedures and those of the installed equipment. Therefore, an independent conclusion cannot be reached regarding the extent to which the units are similar, and the validity of the tests as evidence of qualification has not been established.

b. The Guidelines require that the temperature/pressure profile during the test envelop the expected service conditions for a time duration equivalent to the period from the initiation of the accident until the service conditions return to normal values. This requirement is considered to be essentially satisfied, even though there were some deviations.

c. The fact that the terminal blocks installed in the plant are enclosed within vented junction boxes, while those tested in Reference 2.10 were fully exposed, does not eliminate the need to include the chemical spray environment in the test programs. Experience has shown that deposits of chemicals and contaminants in the spray solution generally are present following test of terminal blocks that are enclosed and that these deposits sometimes contribute to electrical failures. Also, if the terminal blocks are used for signals from electrical transmitters, the presence of moisture, high temperature, chemical spray solution, and nuclear radiations may degrade the signal's accuracy.

d. Contrary to the statement in Reference 2.10, filled phenolics often are strongly susceptible to both thermal and radiation aging (see Appendix C of the Guidelines). Neither thermal nor radiation aging was addressed in the test programs, nor was the large radiation dose associated with a LOCA event. Also, the contribution to the total dose from beta radiation may be significant. The period of qualified life must be established.

LICENSEE RESPONSE:

[No response provided.]

FRC EVALUATION:

FRC makes the following additional comments to support the conclusion presented below:

1. It has not been shown, either by test or analysis, that terminal block failure is unlikely under the effects of thermal aging, radiation, and steam/chemical spray environments postulated to follow a LOCA event. Also, the Licensee has not stated whether the blocks are exposed or installed within junction boxes and whether the presence of moisture could affect the accuracy of instrumentation signals carried by the blocks.

2. The Guidelines require that equipment must be qualified to integrated nuclear radiation dose levels that (i) reflect the sum of both the normal operating dose (for the qualified life period as a minimum) and the accident dose level, and (ii) takes into account the effects of beta radiation and the proximity of the installed equipment to the sump or other concentrated sources of radiation. In reviewing terminal block qualification data referenced in connection with the Palisades plant, FRC noted that the Westinghouse statement regarding radiation qualification was quoted out of context, and that the situation is unsatisfactory for the long term following a LOCA.

3. Aging degradation has not been addressed as required by the Guidelines. The Licensee should evaluate the susceptibility of the terminal blocks to degradation as a result of exposure to temperature and nuclear radiation during the installed life in the plant. If significant degradation is expected to occur, aging must be addressed in the test program and an explicit determination made of qualified life.

4. FRC has reviewed several references which provide statements concerning aging and irradiation effects on the materials used in terminal blocks. It has been stated that the material (wood-flour-filled phenolic) is capable of withstanding continuous service at 125°C. It has also been stated that extrapolated 40-year life temperature ranges from 105°C to 110°C. Other reports indicate that mechanical properties begin to degrade at 0.5 Mrd and that elongation and impact strength are reduced by 25% at 3 to 8 Mrd.

   The mechanical and thermal properties of wood-flour-filled phenolics are highly variable as shown in Appendix F. The data reviewed for the EEQ program demonstrate that data scatter on thermal aging is wide (e.g., 171 hours at 150°C = 40 years, 160 hours at 136°C = 40 years, 100 hours at 126°C = 11.4 years). FRC considers that meaningful forecasts of lifetime and uniform standards for aging damage have not been established for the wood-flour-filled phenolics.

5. With regard to spray, FRC has reviewed 24-hour tests in which deposits accumulated along mold lines of terminal blocks and grounded a terminal. Examination of various terminal blocks after simulated LOCA with chemical spray has indicated conductive deposits on block surfaces that resulted in reduced insulation resistance without complete grounding or short circuit. The Licensee has not analyzed

the effect of high conductivity on instrument signals. Merely maintaining voltage does not assure reliable transmission of level/pressure information.

FRC has also reviewed Sandia Report Number SAND80-2447A presented at the Eighth Water Reactor Safety Research Information meeting held at the National Bureau of Standards from October 27 to 31, 1980. The following statement is presented verbatim from page 1 of the report:

Otmar M. Steutzer
Sandia National Laboratories
Albuquerque, New Mexico 87185

Wire connections in reactor systems are generally made by means of Terminal Blocks (TBs), small insulating boards, each accommodating from 6 to 12 screwdown metal terminals. Figure 1 shows the three models of TBs used in the containment of Three Mile Island, Unit 2 (TMI-2). The blocks are shielded from dirt, or direct steam impingement, by protective enclosures or circuit boxes, many of them similar to the standard fuse boxes. The enclosures are not hermetically sealed and are equipped with breathers or "weep-holes," which at TMI-2 are 6 mm in diameter, but in some other reactors are 25 mm wide. During a steam outbreak, steam can therefore reach the TBs by diffusing through these openings. This makes the insulator surface more conductive. Figure 2 indicates what happens: increased leakage currents (from terminal-to-ground or to another terminal), noise in the circuits, and potentially total electrical breakdown.

TBs have been suspect for a long time. At the urging of the NRC, TBs in safety-related (1E) circuits were replaced in most reactors by splices. At TMI, 620 terminals were eliminated, but there are still 2700 in the containment. And in the case of an accident even non-safety circuits may be important.

The report presents data and a statistical evaluation of results for probability of failure as a function of time and voltage.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because there is no assurance that the terminal blocks would perform reliably or transmit reliable instrument signals under LOCA conditions.

4.6.8 Equipment Item No. 56
Solenoid Valve Located in the Reactor Building
ASCO, Model Not Stated
Nitrogen System Valve (V-23-20)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These are normally closed containment isolation valves that will not change position given a failure of the solenoid valve. They are in a non-harsh temperature/pressure environment and the expected one-year integrated radiation exposure is on the order of 0.1 Mrads. This is below the level at which any detrimental effects will occur.

Based upon the above discussion, there is no reason to believe these valves will not stay closed.

This equipment will either be qualified or replaced by July 1, 1982.

FRC EVALUATION:

The Licensee has not provided, and FRC has found no other source of, valid qualification documentation for this solenoid valve. Therefore, qualification has not been established in accordance with the requirements of the Guidelines.

FRC's review of the Licensee's justification for continued plant operation (Chapter 7 of Reference 1) for this equipment item is given in Appendix D of this report. As noted in Appendix D, the Licensee has not addressed the need for this valve to function in the long-term, post-accident period. The Licensee has not provided any analyses to support the assertion that expected radiation exposure (2.29 Mrd stated on the SCEW sheets) "is below the level at which any detrimental effects will occur." The Licensee should proceed with preventive maintenance activities on an expedited schedule. The manufacturer should be consulted to obtain recommended replacements for coils and other non-metallic components used in these valves.

DELETED MATERIAL IS PROPRIETARY INFORMATION

FRC CONCLUSION:

This equipment is assigned to NRC Category V because valid qualification documentation has not been provided. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to a program of equipment qualification or replacement by June 1982.

4.6.9 Equipment Item Nos. 23, 24, and 25
Solenoid Valve Located in the Reactor Building
23: ASCO Model WPLB83177 (V-23-15)
24: ASCO Model 831424 (V-23-16)
25: ASCO Model X8031A42 (V-23-19)
Nitrogen System Valves
(Final Licensee References 2.6 and 2.7)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These are normally closed containment isolation valves that will not change position given a failure of the solenoid valve. They are in a non-harsh temperature/pressure environment and the expected one-year integrated radiation exposure is on the order of 0.1 Mrads. This is below the level at which any detrimental effects will occur.

Based upon the above discussion, there is no reason to believe these valves will not stay closed.

FRC EVALUATION:

The references cited by the Licensee are not adequately identified and copies were not provided for review. Also, FRC is not aware of valid qualification documentation for this solenoid valve from other sources. Therefore, qualification has not been established in accordance with the requirements of the Guidelines.

FRC's review of the Licensee's justification (Chapter 7 of Reference 1) for this equipment item is given in Appendix D of this report. As noted in Appendix D, the Licensee has not addressed the need for these valves to function during the long-term, post-LOCA period. The Licensee has not provided any analyses to support the assertion that the expected radiation exposure (0.29 Mrd on the SCEW sheets) "is below the level at which any detrimental effects will occur." The Licensee should proceed with preventive maintenance activities on an expedited schedule. The manufacturer should be consulted for recommended replacements for coils and elastomer parts used in these valves.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because valid qualification documentation has not been provided. The solenoid valve should either be qualified or replaced as in the case of Equipment Item No. 56.

4.6.10 Equipment Item 12D
Pressure Switches Located in the Reactor Building
Barton Model 288A
Isolation Condenser Pressure Switches (IB-05-A1, -A2; IB-05-B1, -B2; IB-11-A1, -A2; IB-11-B1, -B2)
(Final Licensee References 2.7 and 2.10)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These switches are provided to sense a sudden pressure change in the Emergency Condenser system following a break in the emergency condenser line. The switches, after sensing the pressure change, will also isolate the Emergency Condenser system (closure of the isolation valves). The initiation of the isolation valve closure takes place 40 seconds (35 seconds + 5) after the line break. The time delay is provided to avoid a spurious trip due to a pressure surge when the Emergency Condenser is put into service under a normal operation. Once a signal for valve closure is initiated by the switches, the valve will complete its closure regardless of the availability of these pressure switches. Therefore, a failure of the switches after the initial 40 seconds will not prevent a closure of the isolation valves. Our analysis indicates that the peak temperature at the valve location at 40 seconds following the break is 170°F. The test report obtained from the switch manufacturer shows that the switch (Barton 288A) did not experience malfunction or physical damage at a test temperature of 212°F. Further, the radiation test given in the test report indicates that the switch operated normally after a radiation exposure of 3 Mrads. Our analysis shows that these switches could experience up to 0.39 Mrads after a full year of radiation exposure due to the accident, based on an extremely conservative assumption of 100% fuel failure in the reactor core.

FRC EVALUATION:

The Licensee has referenced qualification documentation, but did not provide copies for review.

Because the switch provides an important function, qualification documentation is necessary. The Licensee should provide additional information to demonstrate that the possible failure of the switch will not degrade the associated safety-related electrical circuit when the switch is exposed to the harsh conditions of a MSLB/HELB outside the drywell containment.

The Licensee should also estimate the qualified life of the switch and analyze maintenance surveillance records for any abnormal behavior which might limit qualified life.

Qualification of the switch requires that the switch be shown to be operable for one hour plus its normal operating time.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because qualification documentation was not submitted by the Licensee for review. The Licensee should furnish a statement with supporting documentation on qualified life in accordance with Section 4.1.3.

4.6.11 Equipment Item No. 33
Position Switches Located in the Steam Tunnel
Snaplock (NAMCO) Model SL3-C58W
MSIV Position Indicators (NS-04A-1, -2; NS-04B-1, -2)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

The MSIV solenoid valves are used to direct instrument air to hold open the outside containment main steam isolation valves. The MSIV position indication switches are utilized to provide a scram signal when the MSIVs are less than 90% open.

A loss of power or air to the MSIV solenoids causes the MSIVs to fail in the safe direction, closed. Also, redundant protection is provided by the inside containment isolation valves that would not be affected by the environment created by outside containment breaks.

In the event the outside containment MSIV position switch did not provide a scram signal, two scram signals would still be available to ensure the reactor was shut down immediately for a main steam line break. These two signals are the MSIV position switch signal from the inside valves and the reactor low water level signal, both of which would not be affected by the harsh environment created during this event.

The one-year integrated accident exposure of these components is at least two orders of magnitude below that which would cause any degradation.

Based upon the above discussion, it is expected that the main steam isolation function and reactor scram function required to mitigate outside containment will be accomplished.

FRC EVALUATION:

Because it provides an important safety-related function, the switch must be qualified for a MSLB in the steam tunnel. The Licensee did not cite any qualification reference that would demonstrate operability of the switch under accident conditions. The limit switch is required to operate for the short-term period of a postulated MSLB in the steam tunnel. Because in the event of a small break a harsh temperature condition could exist for an extended time period, it is necessary to demonstrate qualification for a minimum of 1 hour.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

These switches are assigned to NRC Category V because no documentation has been provided to support qualification. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to equipment qualification or replacement by June 1982.

4.6.12 Equipment Item No. 55 (previously designated I12)
Relief Valve Operator Located Within the Drywell
Dresser Model 1525 VX (previously shown as General Electric equipment)
Power Operated Relief Valves
(Licensee Reference 2.1)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.8):

Licensee Reference 2.1 describes a steam exposure test performed on a Dresser Electromatic Relief Valve. FRC has the following comments with regard to this reference:

a. It is not evident that the equipment installed in the plant is the same as the item tested. The Guidelines require that the test specimen be the same as the equipment being qualified. The Licensee did not present an analysis comparing the impact of deviations between the test specimen's specific design features, materials, and production procedures and those of the installed equipment. Hence, the validity of the cited test as evidence of qualification has not been established.

b. Although the temperature/pressure profile in the test chamber enveloped the service conditions for an adequate time duration, the test did not include a chemical spray exposure. Because the effects of added moisture and chemical residues may be more damaging than steam alone, FRC concludes that the absence of the chemical spray environment is a potentially serious deficiency.

c. The Guidelines require that radiation exposure should be applied during the test sequence concurrent with, or prior to, the steam exposure, unless it is known that the device contains materials that are not subject to degradation by nuclear radiations. The materials used in this item have not been so identified. FRC concludes that degradation due to irradiation of this item must be addressed, preferably by a test involving simultaneous exposures to steam, chemical spray, and gamma radiation in order that the effects of gamma, heating, and other insulation stresses be accurately simulated.

d. Aging degradation has not been considered, nor has the qualified life been established, nor has a program to ascertain whether any in-service failures during the installed life of the equipment are the result of aging degradation, as are required by the Guidelines.

LICENSEE RESPONSE:

Note C of SCEW Sheet I-9 states that this equipment will either be replaced or qualified by July 1, 1982.

FRC EVALUATION:

The Licensee has provided no additional qualification information. Hence, the comments of the DITER still apply.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because evidence of qualification has not been provided. It is noted that the Licensee plans to qualify or replace this equipment by July 1, 1982.

4.6.13 Equipment Item No. 21B
Solenoid Valves Located in the Reactor Building
ASCO Model 83148
Emergency Condenser Makeup Valves (V-11-34 and V-11-36)
(Licensee Reference 2.11)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These valves provide makeup to the isolation conde..sers. With the minimum water level permitted by technical specifications, the emergency condensers will be available to remove heat at their design capacity without uncovering the heat exchanger tubes for 1 hour 40 minutes with both condensers available and 45 minutes if only one condenser is available.

The emergency condenser system is one of the methods available to control reactor pressure and cool down the plant following a HELB. Since the emergency condenser line break is the break that causes the harsh environment, it is likely that one of the alternate cooldown methods would be utilized.

In the area of the emergency condensers, there are temperature detectors that will detect leaks in the emergency condenser system and annunciate this in the control room. By procedure, the control room operator would isolate the affected system before a rupture developed. Therefore, the actual temperature/pressure environment would not reach the levels indicated in the worst-case analysis.

The one-year integrated radiation exposure to those components is in the order of 0.5 Mrads and an evaluation shows that there will be no detrimental effects with exposure of up to 1 Mrads.

Based on the above discussion, it is evident that the ability to achieve cold shutdown will not be adversely affected by the potential environmental effects of these components. Our evaluation of the component materials revealed that this component contains thermal aging and radiation-sensitive materials (Buna-N and/or fish paper). Therefore, the sensitive component materials will be replaced by June 1982.

FRC EVALUATION:

The reference cited by the Licensee is not adequate to demonstrate qualification. Also, FRC is not aware of qualification documentation for this solenoid valve from other sources. Therefore, qualification has not been established in accordance with the requirements of the Guidelines.

A review of the Licensee's justification (Chapter 7 of Reference 1) for continued plant operation with this equipment item is given in Appendix D of this report.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because valid qualification documentation has not been provided. Although the Licensee's evaluation of this equipment item has not been completed, the Licensee has committed to equipment qualification or replacement by June 1982.

4.6.14 Equipment Item No. 10
Pressure Switches Located in the Reactor Building
General Electric MAC Model 552
Core Spray Pressure Switches (RV-26A,B; RV-40A,C)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

Evaluation of the HELB indicates that two switches (RV-40B and RV-40D) will remain in the ambient temperature and atmospheric pressure conditions (77°F and 15 psia). Evaluation of the component material shows that the material having the most susceptibility to radiation is sheet fiber (fish paper). According to the DOR Guidelines, the radiation susceptibility threshold value of this material is in the order of 0.1 rad, which is the same order of magnitude that these switches may experience after a full year of continuous exposure from the reactor coolant containing 100% of noble gas, 50% of halogen, and 1% of others. Therefore, due to the extremely conservative assumptions used in the radiation analysis, we believe that these switches will function following a HELB. Therefore, six of the ten core spray pressure switches provided (see Items 8 and 12) will be available after a worst-case HELB enabling core spray system to function properly. [This equipment item comes from the four units which are in harsh environment. The SCEW sheet notes that this equipment will either be replaced or qualified by July 1, 1982.]

FRC EVALUATION:

Because the switches provide an important function, qualification documentation is necessary and the Licensee should provide information to demonstrate that the possible failure of the switch will not result in the degradation of the associated safety-related electrical circuit when the switch is exposed to the harsh conditions of a MSLB/HELB outside the drywell containment.

The Licensee should also estimate qualified life and analyze maintenance surveillance records for any abnormal behavior which might limit qualified life.

The Licensee stated that satisfactory system initiation function would occur with 8 of [illegible] drywell containment switches remaining operable. Because the Licensee has not provided electrical system diagrams, FRC could not confirm that the system response would not be degraded.

The Licensee stated that the switch will either be replaced or qualified by July 1, 1982.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because there is no evidence of qualification. The Licensee has committed to qualify or replace this equipment by July 1, 1982.

4.6.15 Equipment Item No. 12C
Level Switches Located in the Reactor Building
Barton Model 288A
Reactor Water Level Switches (RE-18-A through -D)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

The RE-18 switches provide a low-low-low (triple low) signal to the automatic depressurization circuit. This signal could be necesssary if there were a small break that required a rapid depressurization in order to permit a core spray injection. The breaks that cause the harsh environment for these switches do not require the use of the automatic depressurization system. It should be noted that, regardless of the condition of the RE-18 switches, the electromatic relief can be manually initiated by the control room operator if he desires to use them for a cooldown.

The RE-05 switches provide a reactor high-pressure scram signal and control room water-level indication. They are redundant and physically separated.

Another important consideration in evaluating the potential failure of these components due to HELBs is the fact that both the emergency condenser and cleanup line areas are monitored by area temperature detectors. These detectors will warn the control operator of leaks in those systems long before the pipes rupture. This will enable the operator to isolate the leak before the harsh environment is established.

The one-year integrated accident radiation exposure these components might see is about one order of magnitude less than the level that might cause an adverse effect on the most sensitive material used. [The SCEW sheet states that this equipment will be qualified or replaced by July 1, 1982.]

FRC EVALUATION:

Because the switch provides an important function, qualification documentation is necessary. The Licensee should also provide information to demonstrate that the possible failure of the switch will not result in the degradation of the associated safety-related electrical circuit when the switch is exposed to the harsh conditions of a MSLB/HELB outside the drywell containment.

The Licensee should also estimate qualified life and analyze maintenance surveillance records for any abnormal behavior which might limit qualified life.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because there is no evidence of qualification. The Licensee has committed to qualify or replace this equipment by July 1, 1982.

4.6.16 Equipment Item Nos. 14A and 15
Pressure Switches Located in the Reactor Building
14A: Barksdale Model B2T-A12SS
Core Spray Pressure Switches (RE-17-A through -D)
15: Barksdale Model E2T-M12SS
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These pressure switches monitor reactor pressure and are interlocked with the core spray auto initiation logic to prevent core spray injection valves from opening until reactor pressure is below 285 psig.

The core spray system consists of two redundant single failure-proof, low pressure core spray systems. The two postulated HELBs that would create a harsh temperature environment in these areas are cleanup line rupture and emergency condenser line ruptures. Both of these postulated breaks do not require the initiation of core spray for ECCS purposes. In these scenarios, the core spray would be utilized by the control room operator as a safety grade safe shutdown system for reactor water makeup if no high pressure means were available. In that case, these switches are not necessary since the remote manual operation of the core spray injection valves will not be affected by the condition of these switches. The integrated one-year exposure of these components under accident conditions is on the order of 10 Krads; this is significantly below the level of 0.1 Mrads that would cause any detrimental effects on the most sensitive material used in them.

Based on the above considerations, it is expected that the core spray system will be able to perform its ECCS function for inside containment breaks and also provide a safety grade reactor makeup capability for cleanup or emergency condenser line breaks outside containment. [The SCEW sheet notes that this equipment will either be replaced or qualified by July 1, 1982.]

FRC EVALUATION:

The area where the safety-related switches are located is relatively mild, except for radiation exposure, for the accident conditions that the switch is designed to mitigate. The Licensee has not addressed the question of whether these switches could incorrectly indicate a pressure of less than 285 psig under HELB conditions. It is not clear that such an occurrence is a "safety failure."

Because the switch provides an important function, qualification documentation is necessary, and the Licensee should provide information to demonstrate that the possible failure of the switch will not result in the degradation of the associated safety-related electrical circuit when the switch is exposed to the harsh conditions of a MSLB/HELB outside the drywell containment.

The Licensee should also conservatively estimate the qualified life and analyze maintenance surveillance records for any abnormal behavior which might limit qualified life.

The Licensee stated that the switch would either be qualified or replaced by July 1, 1982.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because there is no evidence of qualification. The Licensee has committed to qualify or replace the equipment by July 1, 1982.

4.6.17 Equipment Item No. 18
Level Switches Located in the Reactor Building
Yarway Model C2337
Reactor Water Level Switches (RE-02-A through -D)
(Final Licensee References 2.11 and 2.12)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These switches provide an auto start signal to core spray, a containment isolation signal, a reactor isolation signal, and one of the signals required for an automatic containment spray start.

These switches are redundant and physically separated. Another important consideration in evaluating the potential failure of these components due to HELBs is the fact that both the emergency condenser and cleanup line areas are monitored by area temperature detectors. These detectors will warn the control room operator of leaks in those systems long before the pipes rupture. This will enable the operator to isolate the leak before the harsh environment is established.

The one-year integrated accident radiation exposure these components might see is about one order of magnitude less than the level that might cause an adverse effect on the most sensitive material used.

Based on the above discussion, it is expected that the safety function required by these switches will be accomplished for the postulated HELBs outside containment that create the harsh environment.

Reactor Water Level Switches and Reactor Water Level Switches/Transmitters: RE-18-A through RE-18-D, RE-05-19A, and RE-05-19B

The RE-18 switches provide a low-low-low (triple low) signal to the automatic depressurization circuit. This signal could be necesssary if there was a small break that required a rapid depressurization in order to permit a core spray injection. The breaks that cause the harsh environment for these switches do not require the use of the Automatic Depressurization System. It should be noted that, regardless of the condition of the RE-18 switches, the electromatic relief can be manually initiated by the control room operator if he desires to use them for a cooldown.

The RE-05 switches provide a reactor high-pressure scram signal and control room water-level indication. They are redundant and physically separated.

Another important consideration in evaluating the potential failure of these components due to HELBs is the fact that both the emergency condenser and cleanup line areas are monitored by area temperature detectors. These detectors will warn the control operator of leaks in those systems long before the pipes rupture. This will enable the operator to isolate the leak before the harsh environment is established.

The one-year integrated accident radiation exposure these components might see is about one order of magnitude less than the level that might cause an adverse effect on the most sensitive material used.

FRC EVALUATION:

The Licensee has referenced qualification documents, but did not make them available for review. The Licensee has stated that the documents will be made available to provide evidence of qualification.

Because the switch provides an important function, qualification documentation is necessary, and the Licensee should provide information to demonstrate that the possible failure of the switch will not result in the degradation of the associated safety-related electrical circuit when the switch is exposed to the harsh conditions of a MSLB/HELB outside the drywell containment.

The Licensee should also estimate qualified life and analyze maintenance surveillance records for any abnormal behavior which might limit qualified life.

The Licensee has stated that the equipment will be replaced or will be qualified by July 1, 1982. It should be noted that qualification requirements state that the switch should be qualified for at least one hour plus its normal safety-related operational time.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because evidence of qualification has not been provided. The Licensee has committed to qualify or replace this equipment by July 1, 1982.

4.7 NRC Category VI
EQUIPMENT FOR WHICH QUALIFICATION IS DEFERRED

The equipment items in this section have been addressed by the Licensee in the equipment environmental qualification submittals; however, the qualification review has been deferred by the NRC in accordance with criteria presented in Sections 2.2.3 and 2.2.5 of this report.

4.7.1 Equipment Item No. 5
Pressure Switches Located in the Reactor Building
Static-O-Ring Model 12NKA
Drywell Hi-Pressure Scram Switch (RE-04A through RE-04D)
(Final Licensee References 2.6 and 2.7)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These switches are installed just outside the Drywell wall, and the peak temperature and pressure to be experienced by these switches are 230°F and 16 psia following an emergency condenser break outside the Drywell. However, these switches are installed to monitor the pressure inside the Drywell and are not required to mitigate a HELB outside the Drywell.

FRC EVALUATION:

FRC agrees with the Licensee's position that the switches will only be exposed to a mild environment for any accident that they are designed to mitigate.

Integrated radiation exposures are as high as 1.5 Mrads and could be significant, but an evaluation could not be made because the Licensee did not submit the references for review. In addition, no qualified life assessment in accordance with Section 4.1.3 was presented for this equipment.

FRC CONCLUSION:

This pressure switch is assigned to NRC Category VI because it is believed by the Licensee to be located in a nonharsh area for the accident condition that it is designed to mitigate. Its review is therefore deferred until after February 1, 1981 as discussed in Section 2.2.3. The Licensee should make its references available for qualification verification.

4.7.2 Equipment Item No. 9
Pressure Switches Located in the Reactor Building
Mercoid Model 9-51/DAW-43-156-R2IE
Core Spray Pressure Switches (RV-29A through D, RV-40B,D)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

RV-29A through D:

Evaluation of the HELB indicates that these switches will remain in the ambient temperature and atmospheric pressure conditions (77°F and 15 psia). Evaluation of the component material shows that the material having the most susceptibility to radiation is sheet fiber (fish paper). According to the DOR Guidelines, the radiation susceptibility threshold value of this material is in the order of 0.1 Mrads, which is the same order of magnitude that these switches may experience after a full year of continuous exposure from the reactor coolant containing 100% of noble gas, 50% of halogen, and 1% of others. Therefore, due to the extremely conservative assumptions used in the radiation analyses, we believe that these switches will function following a HELB.

RV-40B, D:

Evaluation of the HELB indicates that two switches (RV-40B and RV-40D) will remain in the ambient temperature and atmospheric pressure conditions (77°F and 15 psia). Evaluation of the component material shows that the material having the most susceptibility to radiation is sheet fiber (fish paper). According to the DOR Guidelines, the radiation susceptibility threshold value of this material is in the order of 0.1 Mrads, which is the same order of magnitude that these switches may experience after a full year of continuous exposure from the reactor coolant containing 100% of noble gas, 50% of halogen, and 1% of others. Therefore, due to the extremely conservative assumptions used in the radiation analysis, we believe that these switches will function following a HELB. Therefore, six of the ten core spray pressure switches provided (see Items 8 and 12) will be available after a worst-case HELB enabling the core spray system to function properly.

FRC EVALUATION:

The area where the safety-related switches are located is relatively mild, except for radiation exposure, for the accident condition that the switch is designed to mitigate [1]. The review of the switch's qualification is therefore deferred until after February 1, 1981, as discussed in Section 2.2.3.

The Licensee should also estimate qualified life and analyze maintenance records for any abnormal behavior which might limit qualified life.

The Licensee has stated that the switch will either be replaced or qualified by July 1, 1982.

FRC CONCLUSION:

This equipment is assigned to NRC Category VI because it is believed by the Licensee to be located in a nonharsh area for the accident condition that it is intended to mitigate. The review of this equipment is deferred until after February 1, 1981, as discussed in Section 2.2.3. Also, the Licensee should furnish a statement on qualified life in accordance with Section 4.1.3.

4.7.3 Equipment Item No. 12A
Pressure Switches Located in the Reactor Building
Barton Model 288A
Containment Pressure Switches (1P15-A through -D)
(Final Licensee References 2.7 and 2.10)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

The peak temperature and pressure seen by these switches are 200°F and 16 psia, respectively, following an emergency condenser line break outside containment (worst-case HELB). The containment pressure switches are provided to monitor the pressure inside containment and are not required to mitigate a line break outside containment.

Drywell Pressure Switch (RV-46-A through -D) [SCEWS Nos.: 55-58]

The peak temperature and pressure seen by these switches are 230°F and 16 psia, respectively, following an emergency condenser line break outside containment (worst-case HELB). The Drywell (containment) pressure switches are provided to monitor the pressure inside containment and are not required to mitigate a line break outside containment.

FRC EVALUATION:

The area where the safety-related switches are located is relatively mild, except for radiation exposure, for the accident condition that the switch is designed to mitigate [1]. The review of the switch's substantiating qualification is therefore deferred until after February 1, 1981 as discussed in Section 2.2.3.

The Licensee has referenced qualification documents, but they were not made available for review. The Licensee has stated that the documents will be made available to provide evidence of qualification.

The Licensee should also estimate qualified life and analyze maintenance surveillance records for any abnormal behavior which might limit qualified life.

There is a concern, however, that failure of these pressure switches could result in the inadvertent actuation of the containment spray system by the plant operator because of an incorrect signal from an unqualified instrument. Although the use of the spray system could result in a vacuum condition occurring in the drywell, there is no concern as long as the torus vacuum relief valve system is fully qualified, as mentioned in Section 4.1.1. It seems prudent for the Licensee to provide qualified pressure switches to more adequately ensure against possible inadvertent containment spray system actuation.

FRC CONCLUSION:

This equipment is assigned to NRC Category VI because it is believed by the Licensee to be located in a nonharsh area for the accident condition that it is intended to mitigate. The review of this equipment is deferred until after February 1, 1981, as discussed in Section 2.2.3. The Licensee should furnish a statement on qualified life in accordance with Section 4.1.3.

4.7.4 Equipment Item No. 12B
Pressure Switches Located in the Reactor Building
Barton Model 288A
Reactor Isolation Switches (RE-22-A through -H)
(Final Licensee References 2.7 and 2.10)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

Due to their location, these switches will remain in non-harsh environment with respect to temperature (77°F) and pressure (15 psia) following a HELB. The radiation dose at this location after a full year of continuous exposure from the reactor coolant containing 100% of noble gas, 50% of halogen, and 1% of others is less than 61 Krads.

The component material having the most susceptibility to radiation is Viton. According to our reference, this material (Viton) has a radiation susceptibility level of 1 Mrads. Therefore, we believe that these switches will function properly following a HELB.

FRC EVALUATION:

The area where the safety-related switches are located is relatively mild, except for radiation exposure, for the accident condition that the switch is designed to mitigate [1]. The review of the switch's substantiating qualification is therefore deferred until after February 1, 1981, as discussed in Section 2.2.3.

The Licensee has referenced qualification documents, but did not make them available for review. The Licensee has stated that the documents will be made available to provide evidence of qualification.

The Licensee should also estimate qualified life and analyze maintenance surveillance records for any abnormal behavior which might limit qualified life.

FRC CONCLUSION:

This equipment is assigned to NRC Category VI because it is believed by the Licensee to be located in a nonharsh area for the accident condition that it is intended to mitigate. The review of this equipment is deferred until after February 1, 1981, as discussed in Section 2.2.3. The Licensee should furnish a statement on qualified life (see Section 4.1.3).

4.7.5 Equipment Item No. 14B
Pressure Switches Located in the Reactor Building
Barksdale Model B2T-A12SS
Reactor Pressure Switches (RE-03-A through -D)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These pressure switches are the switches used to provide a scram signal on reactor high pressure. This is not the scram signal that would be utilized to shut down the reactor in the event of a rupture of either the Emergency Condenser or the cleanup system. These switches are redundant and physically separated. Also, the most severe temperature conditions are caused by different HELBs for the redundant switches. For normal pressurization transients (MSIV closure, turbine trip, and turbine trip without bypass valves), these switches would carry out their safety function almost immediately (less than 60 seconds). Another important consideration in evaluating the potential failure of these components due to HELBs is the fact that both the emergency condenser and cleanup line areas are monitored by area temperature detectors. These detectors will warn the control operator of leaks in those systems long before the pipes rupture. This will enable the operator to isolate the leak before the harsh environment is established.

The one-year integrated accident radiation exposure these components might see is about one order of magnitude less than the level that might cause an adverse effect on the most sensitive material used.

Based upon the above discussion, it is expected that the reactor will scram the postulated rupture of the cleanup system or emergency condenser system due to a reactor low water signal. Also, the ability to scram the reactor on pressurization transients will not be impeded or prevented. [The SCEW sheet notes that this equipment will be qualified or replaced by July 1, 1982.]

FRC EVALUATION:

The area where the safety-related switches are located is relatively mild, except for some radiation exposure, for the accident condition that the switch is designed to mitigate [1]. The review of the switch's substantiating qualification is therefore deferred until after February 1, 1981, as discussed in Section 2.2.3.

Because the switch provides an important function, qualification is necessary for the environment to which the equipment is exposed. The Licensee should provide information to demonstrate that the possible failure of the switch will not result in the degradation of the associated safety-related electrical circuit when the switch is exposed to the harsh conditions of a MSLB/HELB outside the drywell containment.

The Licensee should also conservatively estimate qualified life and analyze maintenance surveillance records for any abnormal behavior might limit qualified life.

The Licensee has stated that the switch will either be replaced or qualified by July 1, 1982.

FRC CONCLUSION:

This equipment is assigned to NRC Category VI because it is believed by the Licensee to be located in a nonharsh area for the accident condition which it is intended to mitigate. The review of this equipment is deferred until after February 1, 1981, as discussed in Section 2.2.3. At that time, the Licensee should provide the references necessary to justify qualification of the equipment. Also, the Licensee should furnish a statement on qualified life in accordance with Section 4.1.3.

4.7.6 Equipment Item No. 16
Pressure Switches Located in the Reactor Building
Barksdale Model B2T
Reactor Pressure Switches (RE-15-A through -D)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These pressure switches are utilized to automatically trip all recirculation pumps and initiate emergency condensers on a reactor high pressure signal. These switches are redundant and physically separated.

Also, the most severe temperature conditions are caused by different HELBs for the redundant switches. For normal pressurization transients (MSIV closure, turbine trip, and turbine trip without bypass valves), these switches would carry out their safety function almost immediately (less than 60 seconds). Another important consideration in evaluating the potential failure of these components due to HELBs is the fact that both emergency condenser and cleanup line area are monitored by area temperature detectors. These detectors will warn the control operators of the leaks in those systems long before the pipes rupture. This will enable the operator to isolate the leak before the harsh environment is established.

The one-year integrated accident radiation exposure these components might see is about one order of magnitude less than the level that might cause an adverse effect on the most sensitive material used.

Based on the above discussion, it is reasonable to assume these components would function if needed for a pressurization transient.

FRC EVALUATION:

The area where the safety-related switches are located is relatively mild, except for radiation exposure, for the accident condition the switch is designed to mitigate [1]. The review of the switch's substantiating qualification is therefore deferred until after February 1, 1981, as discussed in Section 2.2.3.

Because the switch provides an important function, qualification documentation is necessary and the Licensee should provide additional information to demonstrate that the possible failure of the switch will not result in the degradation of the associated safety-related electrical circuit when the switch is exposed to the harsh conditions of a MSLB/HELB outside the drywell containment.

The Licensee should also estimate qualified life and analyze maintenance surveillance records for any abnormal behavior which might limit qualified life.

The Licensee has stated that the switch will either be replaced or qualified by July 1, 1982.

FRC CONCLUSION:

This equipment is assigned to NRC Category VI because it is believed by the Licensee to be located in a nonharsh area for the accident condition it is intended to mitigate. The review of this equipment is deferred until after February 1, 1981, as discussed in Section 2.2.3. At that time, the Licensee should provide the references necessary to justify qualification of the equipment. Also, the Licensee should furnish a statement on qualified life in accordance with Section 4.1.3.

4.7.7 Equipment Item No. 17
Level Switches Located in the Reactor Building
Yarway Model 4316E
Reactor Water Level Switches (RE-05-A,B)
(Final Licensee References 2.11 and 2.12)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These water level switches, along with RE-05-19A and RE-05-19B, provide the signal to scram the reactor at a low water level. These switches are redundant and physically separated. These switches would carry out their safety function almost immediately (less than 60 seconds). Another important consideration in evaluating the potential failure of these components due to HELBs is the fact that both the emergency condenser and cleanup line areas are monitored by area temperature detectors. These detectors will warn the control room operator of leaks in those systems long before the pipes rupture. This will enable the operator to isolate the leak before the harsh environment is established.

The one-year integrated accident radiation exposure these components might see is about one order of magnitude less than the level that might cause an adverse effect on the most sensitive material used.

Based on the above discussion, it is reasonable to assume these components would function if needed for a pressurization transient.

FRC EVALUATION:

The area where the safety-related switches are located is relatively mild, except for radiation exposure, for the accident condition that the switch is designed to mitigate [1]. The review of the switch's substantiating qualification is therefore deferred until after February 1, 1981, as discussed in Section 2.2.3.

The Licensee has referenced qualification documents, but did not make them available for review. The Licensee has stated that the documents will be made available to provide evidence of qualification.

Because the switch provides an important safety-related function, qualification documentation is necessary, and the Licensee should provide information to demonstrate that the possible failure of the switch will not result in the degradation of the associated safety-related electrical circuit when the switch is exposed to the harsh conditions of a MSLB/HELB outside the drywell containment.

The Licensee should also estimate qualified life and analyze the switch's maintenance surveillance records for any abnormal behavior which might limit qualified life.

The Licensee has stated that the equipment will either be replaced or qualified by July 1, 1982. It should be noted that qualification requirements state that the switch should be qualified for at least one hour plus its normal safety-related operational time.

FRC CONCLUSION:

This equipment is assigned to NRC Category VI because it is believed by the Licensee to be located in a nonharsh area for the accident condition it is intended to mitigate. The review of this equipment is deferred until after February 1, 1981, as discussed in Section 2.2.3. At that time, the Licensee should provide the references necessary to justify qualification of the equipment. Also, the Licensee should furnish a statement on qualified life in accordance with Section 4.1.3.

4.7.8 Equipment Item No. 38
Pressure Switch Located in the Reactor Building
Meletron Model 4201E-3B
Drywell Pressure Containment Isolation Valve Switch (PS-153)
(Final Licensee Reference 2.7)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

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LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

This is a high drywell pressure switch that closes a selected group of isolation valves. Since the function is to isolate the containment in the event of breaks inside containment, it is not required to mitigate any outside containment breaks. It, therefore, does not need to be qualified for this application.

FRC EVALUATION:

The area where the safety-related switches are located is relatively mild, except for radiation exposure, for the accident condition that the switch is designed to mitigate [1]. The review of the switch's substantiating qualification is therefore deferred until after February 1, 1981, as discussed in Section 2.2.3.

The Licensee has referenced qualification documents, but did not make them available for review. The Licensee has stated that the documents will be made available to provide evidence of qualification.

Because the switch provides an important function, qualification documentation is necessary, and the Licensee should provide additional information to demonstrate that the possible failure of the switch will not result in the degradation of the associated safety-related electrical circuit when the switch is exposed to the harsh conditions of a MSLB/HELB outside the drywell containment.

The Licensee should also estimate qualified life and analyze maintenance surveillance records for any abnormal behavior which might limit qualified life.

FRC CONCLUSION:

This equipment is assigned to NRC Category VI because it is believed by the Licensee to be located in a nonharsh area for the accident condition it is intended to mitigate. The review of this equipment is deferred until after February 1, 1981, as discussed in Section 2.2.3. At that time, the Licensee should provide the references necessary to justify qualification of the equipment, especially details of radiation analysis. Also, the Licensee should furnish a statement on qualified life in accordance with Section 4.1.3.

4.7.9 Equipment Item No. 41
Level Switches Located in the Reactor Building
Magnetrol Model SIM3 Group 4
Scram Discharge Valve Level Switches (RD-08-A through RD-08-F)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

These switches provide for alarm, rod block, and a reactor scram on a sensed high water level in the instrument scram discharge volume. These components are located in an area that does not see a harsh temperature and pressure environment. Also, the switches do not provide a primary safety function in the event of HELB inside or outside containment. They do serve to back up the signal that provides the reactor scram (high drywell pressure or low water level). The only possible adverse effect that the failure of this switch might create is to allow a scram reset with a significant level of water in the instrument volume. This would require a deliberate action by the Control Room operator in violation of station emergency procedures.

FRC EVALUATION:

The area where the safety-related level switches are located is mild [1]; the review of the substantiating qualification is therefore deferred until after February 1, 1981.

The Licensee's maintenance records should be reviewed to determine if abnormal difficulties have been experienced with the switch. Because the level switch provides an important function, qualification documentation is necessary. In addition, a statement concerning qualified life should be provided by the Licensee.

FRC CONCLUSION:

This level switch is assigned to NRC Category VI because it is believed by the Licensee to be located in a nonharsh area. Its review is therefore deferred until after February 1, 1981, as discussed in Section 2.2.3.

4.7.10 Equipment Item No. 29
Electric Motors Located in the Reactor Building
General Electric Model 5K828848C7
Core Spray Pumps (NZ-01-A through -D)
(Final Licensee Reference 2.14)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

"Qualified"

FRC EVALUATION:

The Licensee has referenced a General Electric test report [2.14], but the report was not made available for review. The Licensee has stated that the report will be made available to provide evidence of qualification for these pump motors.

It is noted that the core spray pump motor is not expected to be exposed to harsh conditions, except for radiation exposure, because the accident temperature is expected to be less than 100°F, the pressure less than 1 psig, and the radiation-integrated exposure less than 0.56 Mrads. For this reason, this motor's qualification review will be deferred until after February 1, 1981. However, maintenance and analysis records should be reviewed to determine if equipment degradation has been abnormal and to assist in the determination of the equipment's qualified life.

FRC CONCLUSION:

This core spray pump motor is assigned to NRC Category VI because the Licensee believes it is located in a nonharsh area. Its review is therefore deferred until after February 1, 1981, as discussed in Section 2.2.3.

4.7.11 Equipment Item No. 13
Pressure Switches Located in the Reactor Building
Meletron Model 372
MSL Low Pressure Switches (RE 23-A through -D)
(Final Licensee Reference 2.7)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE (EQUIPMENT ITEM ADDED IN REFERENCE 1):

Peak temperature and pressure seen by these switches are 218°F and 23 psia, respectively, following a reactor feedwater line break outside containment. The main steam line low pressure switches are provided to monitor a pressure drop in the main steam line due to a MSLB and initiate a closure of the main steam line isolation valve. Therefore, these switches are provided to detect a MSLB and not to detect a break in the feedwater line. Due to the location (different floor level), these switches are also protected from a MSLB.

FRC EVALUATION:

The Licensee has indicated that these switches are in a mild environment for the accident which they are intended to mitigate. Note that they should be qualified for the worst-case MSLB environment and the environment was not identified. (The SCEW sheet indicates no change.)

The qualification document cited by the Licensee was not made available for review.

The Licensee should estimate qualified life in accordance with Section 4.1.3 and analyze maintenance surveillance records for any abnormal behavior which might limit qualified life.

FRC CONCLUSION:

This switch is assigned to NRC Category VI because the Licensee believes it is in a nonharsh area. Its review is therefore deferred until after February 1, 1981, as discussed in Section 2.2.3.

4.8 SUMMARY OF THE EVALUATION

The following tabulations represent a summary of the results of the equipment environmental qualification evaluation conducted by FRC in accordance with the methodology presented in Section 3.

Table 4-1 summarizes the number of equipment items assigned to each NRC qualification category as a result of the evaluation.

Table 4-2 consists of Equipment Environmental Qualification Summary Forms for each equipment item identifying compliance with the qualification requirements defined in Section 3. The following designations are used:

X = A deficiency with respect to compliance with a Guidelines requirement. Deficiencies result in equipment items being categorized as unqualified or qualification not established.

L = A limiting factor with respect to qualification in that qualified life and aging have not been properly considered by the Licensee.

O = Assignment to an NRC qualification category.

R = Replacement of the equipment is planned by the Licensee.

Table 4-1

NUMBER OF EQUIPMENT ITEMS IN EACH QUALIFICATION CATEGORY

NRC Category No.	Category Definitions	Number of Equipment Items
I.a	Equipment Fully Satisfies All Applicable Requirements for the Life of the Plant	0
I.b	Equipment Does Not Meet All Applicable Requirements; However, Deviations Are Judged Acceptable for the Life of the Plant	0
II.a	Equipment Satisfies All Applicable Requirements With the Exception of Qualified Life	8
II.b	Equipment Satisfies All Applicable Requirements With the Exception of Qualified Life Provided That Specific Modifications Are Made	0
II.c	Equipment Does Not Meet All Applicable Requirements; However, Deviations Are Judged Acceptable With the Exception of Qualified Life	6
III	Equipment is Exempt from Qualification Requirements	1
IV.a	Equipment Has Qualification Testing Scheduled	0
IV.b	Equipment Has High Likelihood of Operability; However, Proper Qualification Documentation Has Not Been Made Available for Review	26
V	Equipment is Unqualified	21
VI	Equipment Qualification is Deferred	11
		73

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Table 4-2

Franklin Research Center, A Division of The Franklin Institute, The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK 02.195	REACTOR TYPE BWR	PLANT NAME OYSTER CREEK	PAGE 1
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Jersey Central Power & Light Company	
SEP PLANTS	DOCKET 50-219	NRC TAC 42524	DATE/ENGINEER 4-13-81 *ga*	

SUMMARY REVIEW	1	2	3A	3B	4A	4B	4C	5	6	7	8A	8B	8C	8D	8E	9	10	11	12A	12B
GUIDELINE REQUIREMENTS.	(DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)																			
EVIDENCE OF QUALIFICATION	X	L	L	L	L	L	X	X	X	X	X	X	X	X	X	X	X	X	X	X
RELATIONSHIP TO TEST SPECIMEN							X													
AGING DEGRADATION EVALUATED		L	L	L	L	L														
QUALIFIED LIFE ESTABLISHED		L	L	L	L	L														
PROGRAM TO IDENTIFY AGING		L	L	L	L	L														
QUAL. FOR STEAM EXPOSURE																				
PEAK TEMPERATURE ADEQUATE																				
PEAK PRESSURE ADEQUATE																				
TEST DURATION ADEQUATE																				
REQUIRED PROFILE ENVELOPED																				
QUAL. FOR SUBMERGENCE																				
QUAL. FOR CHEMICAL SPRAY																				
QUAL. FOR RADIATION			L	L		L														
BETA RADIATION CONSIDERED																				
TEST SEQUENCE																				
TEST DURATION (1 HOUR + FUNCTION)																				
QUANTITY OF EQUIPMENT																				
EQUIPMENT INSPECTED AT SITE																				
QUALIFICATION CATEGORY.	(O — CATEGORY DESIGNATION)																			
I-A. QUAL. FOR PLANT LIFE																				
I-B. QUAL. BY JUDGEMENT																				
II-A. QUAL. FOR < PLANT LIFE		O			O															
II-B. QUAL. PENDING MODIFICATION																				
II-C. QUAL. < PLANT LIFE/FRC REVIEW			O	O		O														
III. EXEMPT FROM QUAL.																				
IV-A. QUAL. TEST SCHEDULE																				
IV-B. QUAL. NOT ESTABLISHED	O						O											O		
V. EQUIP. NOT QUALIFIED									O	O	O	O	O	O	O		O			
VI. QUAL. IS DEFERRED								O								O			O	O
REPLACEMENT SCHEDULE	*						R		*	*	*	*	*	*	*	*	*	*		

Table 4-2 (Cont.)

Franklin Research Center A Division of The Franklin Institute The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK 02.195	REACTOR TYPE BWR	PLANT NAME OYSTER CREEK	PAGE 2
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Jersey Central Power & Ligh	
SEP PLANTS	DOCKET 50-219	NRC TAC 42524	DATE/ENGINEER 4-13-81 ga	

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER																			
	12C	12D	13	14A	14B	15	16	17	18	19	20	21A	21B	22	23	24	25	26	27	28
GUIDELINE REQUIREMENTS.	(DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)																			
EVIDENCE OF QUALIFICATION	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
RELATIONSHIP TO TEST SPECIMEN																				X
AGING DEGRADATION EVALUATED												X	X	X	X	X	X	X	X	X
QUALIFIED LIFE ESTABLISHED										X	X	X	X	X	X	X	X	X	X	X
PROGRAM TO IDENTIFY AGING										X	X	X	X	X	X	X	X	X	X	X
QUAL. FOR STEAM EXPOSURE													X							X
PEAK TEMPERATURE ADEQUATE													X							
PEAK PRESSURE ADEQUATE													X							
TEST DURATION ADEQUATE													X							
REQUIRED PROFILE ENVELOPED													X							
QUAL. FOR SUBMERGENCE																				
QUAL. FOR CHEMICAL SPRAY																				
QUAL. FOR RADIATION										X	X	X	X	X	X			X	X	
BETA RADIATION CONSIDERED																				
TEST SEQUENCE																				
TEST DURATION (1 HOUR + FUNCTION)																				
QUANTITY OF EQUIPMENT																				
EQUIPMENT INSPECTED AT SITE																				
QUALIFICATION CATEGORY.	(O — CATEGORY DESIGNATION)																			
I-A. QUAL. FOR PLANT LIFE																				
I-B. QUAL. BY JUDGEMENT																				
II-A. QUAL. FOR < PLANT LIFE																				
II-B. QUAL. PENDING MODIFICATION																				
II-C. QUAL. < PLANT LIFE/FRC REVIEW																				
III. EXEMPT FROM QUAL.																				
IV-A. QUAL. TEST SCHEDULE																				
IV-B. QUAL. NOT ESTABLISHED										O	O	O		O				O	O	O
V. EQUIP. NOT QUALIFIED	O	O		O		O			O				O		O	O	O			
VI. QUAL. IS DEFERRED			O		O		O	O												
REPLACEMENT SCHEDULE	*	*		*	*	*	*	*	*	*	*	*	*	*	*			*	*	*

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Table 4-2 (Cont.)

Franklin Research Center, A Division of The Franklin Institute, The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK 32195	REACTOR TYPE BWR	PLANT NAME OYSTER CREEK	PAGE 3
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Jersey Central Power & Light	
SEP PLANTS	DOCKET 50-219	NRC TAC 42524	DATE/ENGINEER 4-13-81 ga	

SUMMARY REVIEW	29	30	31A	31B	32A	32B	33	34A	34B	34C	35	36	37	38	39	40	41	42	43	44
GUIDELINE REQUIREMENTS. (DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)																				
EVIDENCE OF QUALIFICATION	X	X	L	X	L	X	X	L	L	X	X	X	X	X		X	X	X	X	X
RELATIONSHIP TO TEST SPECIMEN								L		X						X				X
AGING DEGRADATION EVALUATED			L		L			L	L											
QUALIFIED LIFE ESTABLISHED			L		L			L	L	X	X	X	X			X				X
PROGRAM TO IDENTIFY AGING			L		L			L	L	X	X	X	X			X				X
QUAL. FOR STEAM EXPOSURE										X						X				X
PEAK TEMPERATURE ADEQUATE																				
PEAK PRESSURE ADEQUATE																				
TEST DURATION ADEQUATE																				
REQUIRED PROFILE ENVELOPED																				
QUAL. FOR SUBMERGENCE																				
QUAL. FOR CHEMICAL SPRAY				X		X													X	
QUAL. FOR RADIATION									L	X	X	X	X							X
BETA RADIATION CONSIDERED																				
TEST SEQUENCE																				
TEST DURATION (1 HOUR + FUNCTION)				X		X							X						X	
QUANTITY OF EQUIPMENT																				
EQUIPMENT INSPECTED AT SITE																				
QUALIFICATION CATEGORY. (O — CATEGORY DESIGNATION)																				
IA. QUAL. FOR PLANT LIFE																				
I-B. QUAL. BY JUDGEMENT																				
II-A. QUAL. FOR < PLANT LIFE			O		O			O												
II-B. QUAL. PENDING MODIFICATION																				
II-C. QUAL. < PLANT LIFE/FRC REVIEW									O											
III. EXEMPT FROM QUAL.															O					
IV-A. QUAL. TEST SCHEDULE																				
IV-B. QUAL. NOT ESTABLISHED				O		O	O			O	O	O	O			O		O	O	O
V. EQUIP. NOT QUALIFIED		O																		
VI. QUAL. IS DEFERRED	O													O			O			
REPLACEMENT SCHEDULE										R	*	*		*			*			R

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Table 4-2 (Cont.)

Franklin Research Center A Division of The Franklin Institute The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK 02.195	REACTOR TYPE BWR	PLANT NAME OYSTER CREEK	PAGE 4
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Jersey Central Power & Light	
SEP PLANTS	DOCKET 50-219	NRC TAC 42524	DATE/ENGINEER 4-13-81 ga	

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER											
	45	46	47	48	49	50	51	52	53	54	55	56
GUIDELINE REQUIREMENTS,	(DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)											
EVIDENCE OF QUALIFICATION	X	X	X	X	L	L	X	L	L	L	X	X
RELATIONSHIP TO TEST SPECIMEN	X										X	
AGING DEGRADATION EVALUATED	X			X			X				X	X
QUALIFIED LIFE ESTABLISHED	X	X	X	X	L	L	X	L	L	L	X	X
PROGRAM TO IDENTIFY AGING	X	X	X	X	L	L	X	L	L	L	X	X
QUAL. FOR STEAM EXPOSURE				X								
PEAK TEMPERATURE ADEQUATE												
PEAK PRESSURE ADEQUATE												
TEST DURATION ADEQUATE												
REQUIRED PROFILE ENVELOPED												
QUAL. FOR SUBMERGENCE												
QUAL. FOR CHEMICAL SPRAY	X			X								
QUAL. FOR RADIATION		X	X	X			X				X	X
BETA RADIATION CONSIDERED												
TEST SEQUENCE												
TEST DURATION (1 HOUR + FUNCTION)												
QUANTITY OF EQUIPMENT												
EQUIPMENT INSPECTED AT SITE												
QUALIFICATION CATEGORY,	(O — CATEGORY DESIGNATION)											
IA. QUAL. FOR PLANT LIFE												
I-B. QUAL. BY JUDGEMENT												
II-A. QUAL. FOR < PLANT LIFE					O	O			O			
II-B. QUAL. PENDING MODIFICATION												
II-C. QUAL. < PLANT LIFE/FRC REVIEW								O		O		
III. EXEMPT FROM QUAL.												
IV-A. QUAL. TEST SCHEDULE												
IV-B. QUAL. NOT ESTABLISHED		O	O				O				O	
V. EQUIP. NOT QUALIFIED	O			O								O
VI. QUAL. IS DEFERRED												
REPLACEMENT SCHEDULE	*	*	*								*	*

5. CONCLUSIONS

The tabulations presented in Section 4.8 represent a summary of the results of the equipment environmental qualification (EEQ) assessment conducted by FRC in accordance with the methodology presented in Section 3. The evaluations are based on the available qualification documentation provided by the Licensee, supplemented in several cases by other relevant technical information. The major deficiencies that have been identified are shown in the Equipment Environmental Qualification Summary Forms (Table 4-2). The review has shown that qualification documentation for many equipment items is inadequate or non-existent, and that additional information is essential.

The DOR Guidelines require the Licensee to have ongoing programs to review surveillance and maintenance records in order to assure that safety-related equipment that exhibits age-related degradation be identified and, if necessary, replaced. No evidence of such programs was included in the Licensee submittal.

The Licensee has offered several system-related arguments to exempt certain equipment items from qualification review. Most of these arguments fall into two categories: (1) the backup system redundancy can adequately accomplish the function, or (2) the equipment need only survive for a few minutes in order to accomplish its intended function. The FRC conclusions regarding these arguments are given in Section 4 for each equipment item, and a more detailed analysis is presented in Appendix D.

The present assessment of the status of environmental qualification of the safety-related electrical equipment installed in Oyster Creek involves only equipment located in the "harsh environment" areas and needed to ensure hot shutdown of the plant. The EEQ review of equipment items located in "mild" areas and of equipment needed for TMI Action Plan compliance has been deferred by the Licensee until after February 1, 1981.

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   Limitorque Corp., 21-Aug-80

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   2.8 IE Bulletin 79-01B
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APPENDIX A - ENVIRONMENTAL SERVICE CONDITIONS

The Licensee provided information concerning "harsh" environmental service conditions in various locations of the plant where safety-related equipment is installed: the containment drywell, the reactor building, and the steam tunnel. The EEQ review of equipment needed to achieve cold shutdown status has been deferred in accordance with Section 2.2.5. In addition, the Licensee has deferred the EEQ review of equipment located in "mild areas," as discussed in Section 2.2.3. Therefore, only the "harsh" environments were discussed in Reference 1, and considered in this report.

In Table 1 of Reference 1, the Licensee presented the worst-case temperature/pressure/radiation service conditions that each safety-related equipment item located outside of the containment drywell could experience for different types of postulated accidents. According to the Licensee, the maximum duration of the pressure/temperature excursion is 1200 seconds before conditions return to normal.

Figures A-1, A-2, and A-3 define the results of the containment drywell MSLB analysis, showing the expected temperature and pressure excursions after the worst-case postulated accident.

Environment 1 -- Within Reactor Containment Drywell

Normal Operation

Temperature	70°-135°F (120°F nominal)
Pressure	15.7 psia
Humidity	60% (nominal)
Radiation	(Not stated, included in accident dose)

Accident Conditions

For BWR plants, Section 4.1 of the DOR Guidelines states that the temperature component of the environmental service conditions within the drywell for the loss-of-coolant accident (LOCA) will be 340°F for 6 hours. This exposure is intended as a bounding condition to reflect the superheated steam release associated with the most severe main steam line break (MSLB) accident. Supplement 2 to IE Bulletin 79-01B states that a plant-specific analysis may be used in lieu of 340°F for 6 hours. The Licensee has provided plant-specific analyses for both LOCA and MSLB events. The latter is more severe than the former, and therefore is the basis for establishing the status of qualification. The Licensee has investigated a wide spectrum of postulated break sizes, break locations, and single failures associated with a LOCA or MSLB accident. The NRC has acknowledged that the MSLB accident appears to be the limiting environmental service condition for which equipment located within the containment drywell is to be evaluated.

The environmental parameters associated with the LOCA and MSLB events used for the assessment of qualification of equipment inside the containment are:

Temperature	Figures A-1, A-3 (Ref. 1)
Pressure	Figure A-2 (Ref. 1)
Radiation	57 Mrd (1 year, gamma radiation only; includes 40-year normal operation)
Humidity	100% (assumed)
Flood Level	Not stated
Spray	Demineralized water containing sodium dichromate

Environment 2 -- Within Reactor Building; Outside of Containment Drywell

Normal Operation

Temperature	Not stated
Pressure	Not stated
Radiation	Not stated
Humidity	Not stated

Accident Conditions

Temperature	See Table 1	
Pressure	See Table 1	
Radiation	See Table 1	
Humidity	100% (Assumed)	
Flood Level	Elevation	Depth (Ref. 1)
	-19'6"	2 1/4"
	23'6"	2 7/8"
	51'3"	2 7/16"
	75'3"	1 3/8"
	95'3"	9/16"

Environment 3 -- Steam Tunnel

Normal Operation

Temperature	Not stated
Pressure	Not stated
Radiation	Not stated
Humidity	Not stated

Accident Conditions

Temperature	See Table 1
Pressure	See Table 1
Radiation	See Table 1
Humidity	100% (Assumed)
Flood Level	0, Water will drain to Turbine Bldg. Sump Pump at -8' elev.

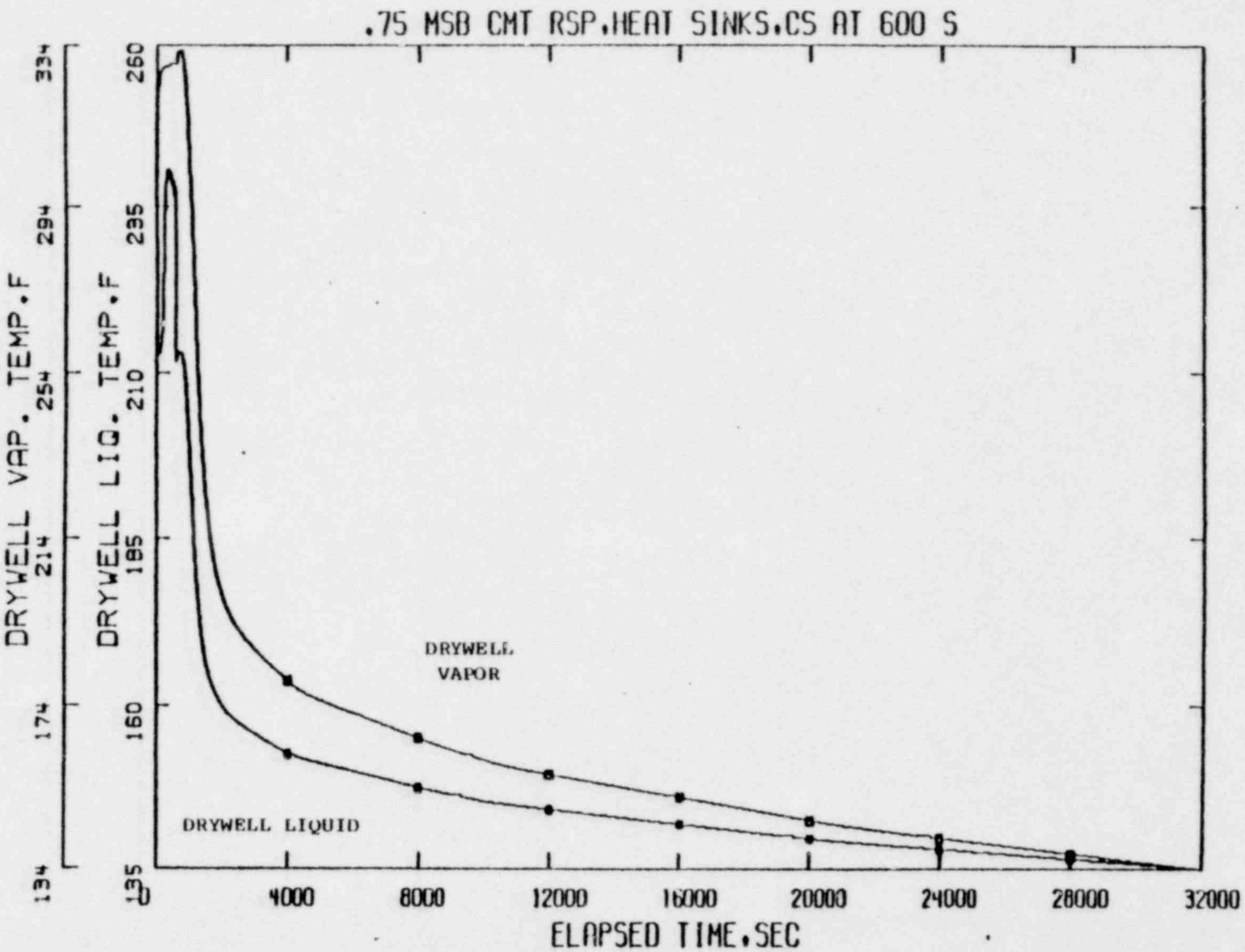

Figure A-1. Long-Term Response to Temperature Versus Time for Postulated Main Steam Line Break in the Drywell [1]

FIGURE SUPPLIED BY THE LICENSEE

ENVIRONMENTAL SERVICE CONDITIONS
TEMPERATURE VERSUS TIME

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ENVIRONMENTAL SERVICE CONDITIONS
PRESSURE VERSUS TIME

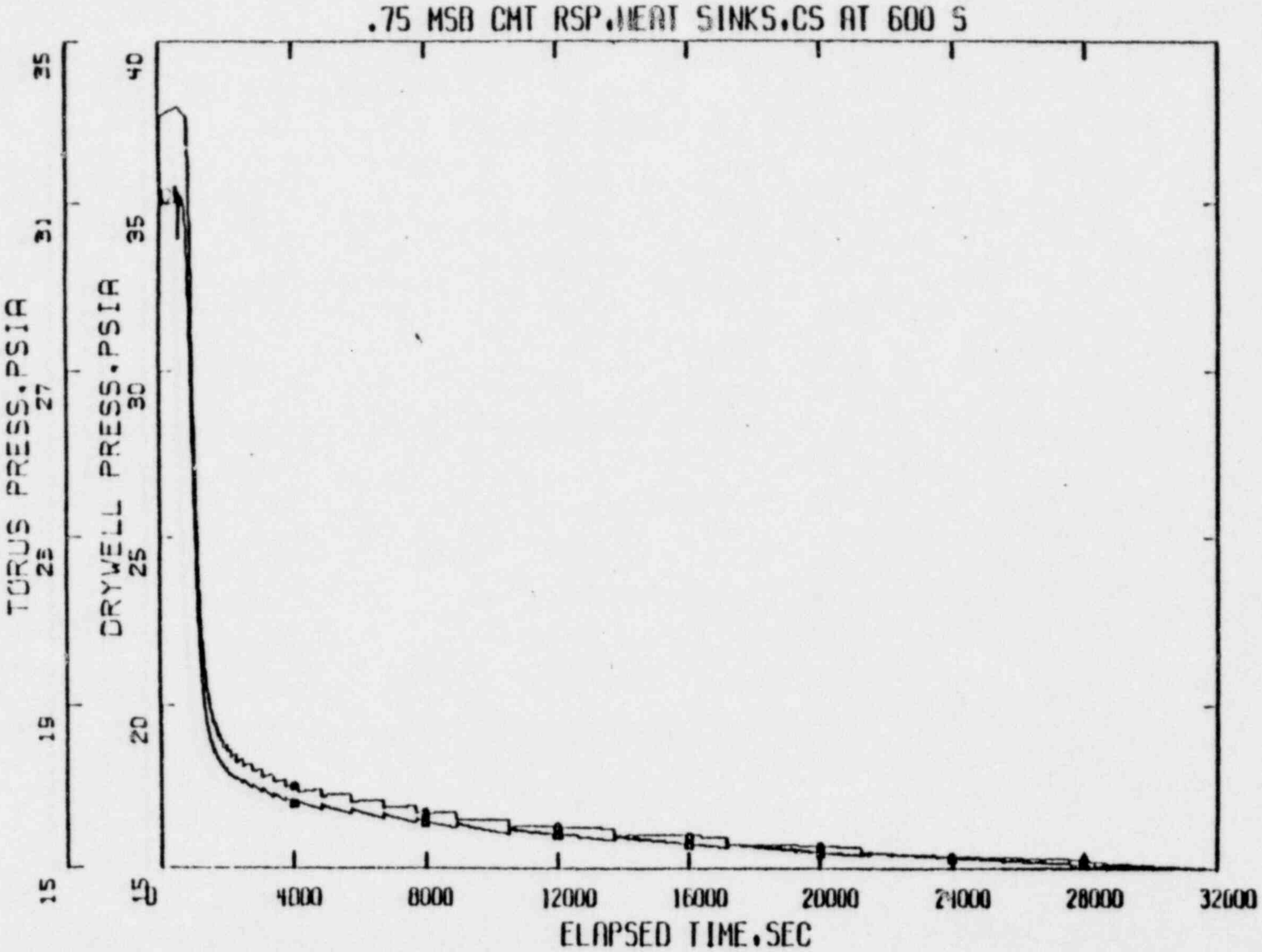

Figure A-2. Pressure Versus Time for Postulated Main Steam Line Break in the Drywell [1]

FIGURE SUPPLIED BY THE LICENSEE

ENVIRONMENTAL SERVICE CONDITIONS
PRESSURE VERSUS TIME

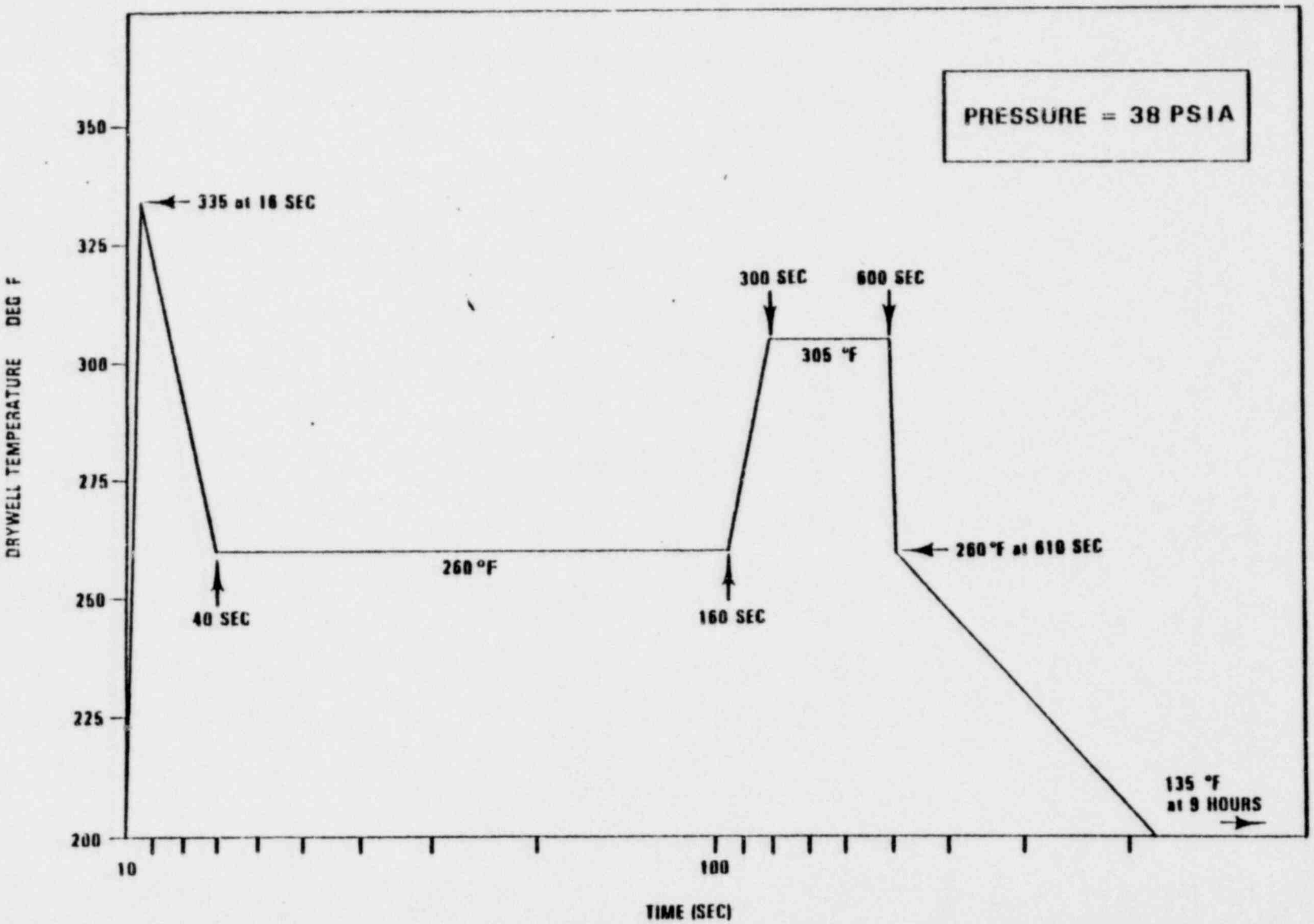

Figure A-3. Temperature Versus Time for Postulated Main Steam Line Break in the Drywell [1]

FIGURE SUPPLIED BY THE LICENSEE

Table A-1. Oyster Creek Nuclear Generating Station
Electrical Equipment Environmental Conditions [1]

*Indicates that equipment is not required to mitigate the consequences of the accident outside of containment or to achieve a safe shutdown for that accident. For a break inside containment, asterisked items are needed to mitigate the accident, however, the environmental conditions for these asterisked items would be normal ambient conditions.

	Target Designation	Description	Location Coordinates	Approximate Elevation	Worst Case Line Break	Peak Temperature (°F)	Temperature Profile	Peak Pressure (PSIA)	Total Integrated Radiation Exposure (1 Yr.) NUREG 0578 Assumptions (RADS)
1	IA 83 A	AIG Pressure Switch	R_4 x East Drywell Wall	65'	Cleanup System	202°	Fig. 10	16 PSIA	3.7×10^4 R
	IA 83 B	AIG Pressure Switch	R_3 x East Drywell Wall	72'	Cleanup System	215°	Fig. 12	16 PSIA	$<6.1 \times 10^4$ R
	IA 83 C	AIG Pressure Switch	RKO 2	72'	Cleanup System	212°	Fig. 11	16 PSIA	$<6.1 \times 10^4$ R
	IA 83 D	AIG Pressure Switch	RKO 1	72'	Emer. Cond.	230°	Fig. 4	16 PSIA	1.4×10^4 R
	IA 83 E	AIG Pressure Switch	RKO 3	55'	Emer. Cond.	230°	Fig. 4	16 PSIA	3.9×10^5 R
2	V 26 16*	Drywell Vent & Purge Valve	R_2 x R_Y	25'	-	77°*	-	15 PSIA*	2.0×10^5 R
	V 26 18*	Drywell Vent & Purge Valve	R_2 x R_Y	25'	-	77°*	-	15 PSIA*	3.0×10^5 R
3	V 21 5*	Containment Spray Valve	R_3 x R_C	33'	Cleanup System	140°*	Fig. 13*	15 PSIA*	5.1×10^5 R
	V 21 11*	Containment Spray Valve	R_5 x Drywell Wall	62'	Cleanup System	202°*	Fig. 10*	16 PSIA*	5.1×10^5 R
4	V 5 167*	Reactor Bldg. Closed Cp Cooling System Valve	R_4 x Drywell Wall	49'	Cleanup System	95°*	-	16 PSIA*	2.5×10^4 R
	V 5 147*	Reactor Bldg. Closed Cp Cooling System Valve	R_4 x Drywell Wall	49'	Cleanup System	95°*	-	16 PSIA*	2.5×10^4 R
5	V 21 13	Containment Spray Valve	R_1 R_2 x R_C; R_D	27'	Cleanup System	140°*	Fig. 13	15 PSIA	6.7×10^5 R
	V 21 17	Containment Spray Valve	R_6 R_7 x R_B R_C	27'	-	77°	-	15 PSIA	3.7×10^5 R
	V 21 3	Containment Spray Valve	Southeast Corner Rm.	19'	Main Steam	165°	Fig. 14	15 PSIA	6.0×10^5 R
	V 21 9	Containment Spray Valve	Northeast Corner Rm.	19'	Main Steam	155°	Fig. 16	15 PSIA	6.1×10^5 R
	V 21 1	Containment Spray Valve	Southeast Corner Rm.	19'	Main Steam	165°	Fig. 14	15 PSIA	1.0×10^6 R
	V 21 7	Containment Spray Valve	Northeast Corner Rm.	19'	Main Steam	165°	Fig. 17	15 PSIA	6.3×10^5 R
	RE 01 A*	Drywell Pressure Scram Switch	℄ Reactor Vessel x R_6 R_7	55'	Emer. Cond.	230°*	Fig. 4*	16 PSIA*	1.5×10^6 R
	RE 01 B*	Drywell Pressure Scram Switch	℄ Reactor Vessel x R_6 R_7	55'	Emer. Cond.	230°*	Fig. 4*	16 PSIA*	1.5×10^6 R
	RE 01 C*	Drywell Pressure Scram Switch	R_1 x South Drywell Wall	55'	Emer. Cond.	230°*	Fig. 4*	16 PSIA*	2.9×10^5 R
	RE 04 D*	Drywell Pressure Scram Switch	R_1 x South Drywell Wall	55'	Emer. Cond.	230°*	Fig. 4*	16 PSIA*	2.8×10^5 R

FIGURE SUPPLIED
BY THE LICENSEE

POOR ORIGINAL

Table A-1. Oyster Creek Nuclear Generating Station
Electrical Equipment Environmental Conditions (Cont.)

Equipt Designation	Description	Location Coordinates	Approximate Elevation	Worst Case Line Break	Peak Temperature (°F)	Temperature Profile	Peak Pressure (PSIA)	Total Integrated Radiation Exposure (1 Yr.) NUREG 0578 Assumptions (RADS
[illegible] A	Reactor Vessel Pressure Trans	RKO 1	72'	Emer. Cond.	330°	Fig. 4	16 PSIA	1.4×10^4 R
[illegible] B	Reactor Vessel Pressure Trans	RKO 2	72'	Cleanup System	315°	Fig. 11	16 PSIA	$<6.1 \times 10^4$ R
[illegible] A	Reactor Vessel Pressure Trans	RKO 1	72'	Emer. Cond.	330°	Fig. 4	16 PSIA	1.4×10^4 R
[illegible] B	Reactor Vessel Pressure Trans	RKO 2	72'	Cleanup System	315°	Fig. 11	16 PSIA	$<6.1 \times 10^4$ R
[illegible]	Isolation Condenser Level Trans	$R_4 \times$ ℄ of "A" Iso. Cond.	98'	Emer. Cond.	310°	Fig. 5	16 PSIA	5.3×10^5 R
[illegible]	Isolation Condenser Level Trans	$R_4 \times$ ℄ of "A" Iso. Cond.	98'	Emer. Cond.	310°	Fig. 5	16 PSIA	5.3×10^5 R
[illegible]	Isolation Condenser Level Trans	$R_4 \times$ ℄ of "B" Iso. Cond.	98'	Emer. Cond.	310°	Fig. 5	16 PSIA	5.3×10^5 R
[illegible]	Isolation Condenser Level Trans	$R_4 \times$ ℄ of "B" Iso. Cond.	98'	Emer. Cond.	310°	Fig. 5	16 PSIA	5.3×10^5 R
[illegible] (16 switches)	Reactor Isolation Temp Switch	Location to be specified by JCP&L						
[illegible] 13A	Reactor Water Level Trans	RKO 1	72'	Emer. Cond.	330°	Fig. 4	16 PSIA	1.4×10^4 R
[illegible] 13B	Reactor Water Level Trans	RKO 2	72'	Cleanup System	315°	Fig. 11	16 PSIA	$<6.1 \times 10^4$ R
[illegible] 13A	Reactor Water Level Trans	RKO 1	72'	Emer. Cond.	330°	Fig. 4	16 PSIA	1.4×10^4 R
[illegible] 13B	Reactor Water Level Trans	RKO 2	72'	Cleanup System	315°	Fig. 11	16 PSIA	$<6.1 \times 10^4$ R
RV 29 A	Core Spray Pressure Switch	NW Corner Room	-19'	-	77°	-	15 PSIA	4.4×10^5 R
RV 29 B	Core Spray Pressure Switch	SW Corner Room	-19'	-	77°	-	15 PSIA	2.6×10^5 R
RV 29 C	Core Spray Pressure Switch	NW Corner Room	-19'	-	77°	-	15 PSIA	4.4×10^5 R
RV 29 D	Core Spray Pressure Switch	SW Corner Room	-19'	-	77°	-	15 PSIA	2.6×10^5 R
RV 40 A	Core Spray Pressure Switch	R_{11} x North Drywell Wall	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	1.7×10^5 R
RV 40 B	Core Spray Pressure Switch	R_2 R_3 x R_E - R_F	27'		77°	-	15 PSIA	2.3×10^5 R
RV 40 C	Core Spray Pressure Switch	R_{11} x North Drywell Wall	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	1.7×10^5 R
RV 40 D	Core Spray Pressure Switch	R_2 R_3 x R_E - R_F	27'	-	77°	-	15 PSIA	2.3×10^5 R
RV 26 A	Core Spray Flow Trans.	RKO 3	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	3.9×10^5 R
RV 26 B	Core Spray Flow Trans.	R_2 R_3 x R_E	81'	Emer. Cond.	315°	Fig. 8	16 PSIA	8.0×10^5 R
IB 06 E	Isolation Cond. Area Temp Det	C.g. Cond. NE 01A	115'	Emer. Cond.	310°	Fig. 5	16 PSIA	$<6.1 \times 10^4$ R
IB 05 F	Isolation Cond. Area Temp Det	C.g. Cond. NE 01B	115'	Emer. Cond.	310°	Fig. 5	16 PSIA	$<6.1 \times 10^4$ R
IB 05 G	Isolation Cond. Area Temp Det	10'S of R1 x Rc Rd	90'	Emer. Cond.	290°	Fig. 6	16 PSIA	1.0×10^6 R
IB 05 H	Isolation Cond. Area Temp Det	10'N of R1 x Rc Rd	90'	Emer. Cond.	290°	Fig. 6	16 PSIA	9.4×10^5 R
IP 93 A	Containment Spray Flow Mtr.	R_C R_{11} x North RB Wall	25'	-	77°	-	15 PSIA	3.4×10^5 R
IP 93 B	Containment Spray Flow Mtr.	R_C R_{11} x South RB Wall	25'	Cleanup System	140°	Fig. 13	15 PSIA	1.7×10^5 R

FIGURE SUPPLIED
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Table A-1. Oyster Creek Nuclear Generating Station
Electrical Equipment Environmental Conditions (Cont.)

[illegible]

Tag Identification	Description	Location Coordinates	Approximate Elevation	Worst Case Line Break	Peak Temperature (°F)	Temperature Profile	Peak Pressure (PSIA)	Total Integrated Radiation Exposure (1 Yr.) NUREG 0588 Assumptions (RAD)
[illegible] A*	Containment Press. Switch	RKO 3	55'	Emer. Cond.	330°*	Fig. 4*	16 PSIA*	3.9×10^5 R
[illegible] B*	Containment Press. Switch	RKO 3	55'	Emer. Cond.	330°*	Fig. 4*	16 PSIA*	3.9×10^5 R
[illegible] C*	Containment Press. Switch	RKO 3	55'	Emer. Cond.	330°*	Fig. 4*	16 PSIA*	3.9×10^5 R
[illegible] D*	Containment Press. Switch	RKO 3	55'	Emer. Cond.	330°*	Fig. 4*	16 PSIA*	3.9×10^5 R
[illegible]	Drywell Press. Trans.	RKO 3	55'	Emer. Cond.	330°*	Fig. 4*	16 PSIA*	3.9×10^5 R
[illegible] A*	Drywell Press. Switch	RKO 3	55'	Emer. Cond.	330°*	Fig. 4*	16 PSIA*	3.9×10^5 R
[illegible] B*	Drywell Press. Switch	RKO 3	55'	Emer. Cond.	330°*	Fig. 4*	16 PSIA*	3.9×10^5 R
[illegible] C*	Drywell Press. Switch	RKO 3	55'	Emer. Cond.	330°*	Fig. 4*	16 PSIA*	3.9×10^5 R
[illegible] D*	Drywell Press. Switch	RKO 3	55'	Emer. Cond.	330°*	Fig. 4*	16 PSIA*	3.9×10^5 R
[illegible] A*	MSL Low Press. Switch	Reactor Fd. Pump Room x North Wall	5'	Reactor Feed	210°*	Fig. 2*	23 PSIA*	$<6.1 \times 10^4$ R
[illegible] B*	MSL Low Press. Switch	Reactor Fd. Pump Room x South Wall	5'	Reactor Feed	210°*	Fig. 2*	23 PSIA*	$<6.1 \times 10^4$ R
[illegible] C*	MSL Low Press. Switch	Reactor Fd. Pump Room x North Wall	5'	Reactor Feed	210°*	Fig. 2*	23 PSIA*	$<6.1 \times 10^4$ R
[illegible] D*	MSL Low Press. Switch	Reactor Fd. Pump Room x South Wall	5'	Reactor Feed	210°*	Fig. 2*	23 PSIA*	$<6.1 \times 10^4$ R
[illegible]	Reactor Isolation Switch	R_F, R_4 x Drywell Wall	27'	-	77°*	-	15 PSIA*	$<6.1 \times 10^4$ R
[illegible]	Reactor Isolation Switch	R_F, R_4 x Drywell Wall	27'	-	77°*	-	15 PSIA*	$<6.1 \times 10^4$ R
[illegible]	Reactor Isolation Switch	R_F, R_4 x Drywell Wall	27'	-	77°*	-	15 PSIA*	$<6.1 \times 10^4$ R
[illegible]	Reactor Isolation Switch	R_F, R_4 x Drywell Wall	27'	-	77°*	-	15 PSIA*	$<6.1 \times 10^4$ R
[illegible]	Reactor Isolation Switch	R_F, R_4 x Drywell Wall	27'	-	77°*	-	15 PSIA*	$<6.1 \times 10^4$ R
[illegible]	Reactor Isolation Switch	R_F, R_4 x Drywell Wall	27'	-	77°*	-	15 PSIA*	$<6.1 \times 10^4$ R
[illegible]	Reactor Isolation Switch	R_F, R_4 x Drywell Wall	27'	-	77°*	-	15 PSIA*	$<6.1 \times 10^4$ R
[illegible]	Reactor Isolation Switch	R_F, R_4 x Drywell Wall	27'	-	77°*	-	15 PSIA*	$<6.1 \times 10^4$ R
[illegible]	Isolation Condenser ΔP Switch	RKO 3	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	3.9×10^5 R
[illegible]	Isolation Condenser ΔP Switch	RKO 3	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	3.9×10^5 R
[illegible]	Isolation Condenser ΔP Switch	RKO 3	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	3.9×10^5 R
[illegible]	Isolation Condenser ΔP Switch	RKO 3	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	3.9×10^5 R
[illegible]	Isolation Condenser ΔP Switch	RKO 3	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	3.9×10^5 R
[illegible]	Isolation Condenser ΔP Switch	RKO 3	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	3.9×10^5 R
[illegible]	Isolation Condenser ΔP Switch	RKO 3	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	3.9×10^5 R
[illegible]	Isolation Condenser ΔP Switch	RKO 3	55'	Emer. Cond.	330°	Fig. 4	16 PSIA	3.9×10^5 R

FIGURE SUPPLIED BY THE LICENSEE

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Table A-1. Oyster Creek Nuclear Generating Station
Electrical Equipment Environmental Conditions (Cont.)

	Target Designation	Description	Location Coordinates	Approximate Elevation	Worst Case Line Break	Peak Temperature (°F)	Temperature Profile	Peak Pressure (PSIA)	Total Integrated Radiation Exposure (1 Yr.) NUREG 0578 Assumptions
22	RE-17-A	Core Spray Press. Switch	RKO-1	72'	Emer. Cond.	230°	Fig. 4	16 PSIA	1.4 x 10[illegible] R
	RE-17-B	Core Spray Press. Switch	RKO-2	72'	Cleanup System	215°	Fig. 11	16 PSIA	<6.1 x 10[illegible] R
	RE-17-C	Core Spray Press. Switch	RKO-1	72'	Emer. Cond.	230°	Fig. 4	16 PSIA	1.4 x 10[illegible] R
	RE-17-D	Core Spray Press. Switch	RKO-2	72'	Cleanup System	215°	Fig. 11	16 PSIA	<6.1 x 10[illegible] R
23	RE-15-A	Reactor Vessel Press. Switch	RKO-1	72'	Emer. Cond.	230°	Fig. 4	16 PSIA	1.4 x 10[illegible] R
	RE-15-B	Reactor Vessel Press. Switch	RKO-2	72'	Cleanup System	215°	Fig. 11	16 PSIA	<6.1 x 10[illegible] R
	RE-15-C	Reactor Vessel Press. Switch	RKO-1	72'	Emer. Cond.	230°	Fig. 4	16 PSIA	1.4 x 10[illegible] R
	RE-15-D	Reactor Vessel Press. Switch	RKO-2	72'	Cleanup System	215°	Fig. 11	16 PSIA	<6.1 x 10[illegible] R
24	RE-03-A	Reactor Pressure Switch	RKO-1	72'	Emer. Cond.	230°	Fig. 4	16 PSIA	1.4 x 10[illegible] R
	RE-03-B	Reactor Pressure Switch	RKO-2	72'	Cleanup System	215°	Fig. 11	16 PSIA	<6.1 x 10[illegible] R
	RE-03-C	Reactor Pressure Switch	RKO-1	72'	Emer. Cond.	230°	Fig. 4	16 PSIA	1.4 x 10[illegible] R
	RE-03-D	Reactor Pressure Switch	RKO-2	72'	Cleanup System	215°	Fig. 11	16 PSIA	<6.1 x 10[illegible] R
25	RE-05-A	Reactor Water Level Switch	RKO-1	72'	Emer. Cond.	230°	Fig. 4	16 PSIA	1.4 x 10[illegible] R
	RE-05-B	Reactor Water Level Switch	RKO-2	72'	Cleanup System	215°	Fig. 11	16 PSIA	<6.1 x 10[illegible] R
26	RE-02-A	Reactor Water Level Switch	RKO-1	72'	Emer. Cond.	230°	Fig. 4	16 PSIA	1.4 x 10[illegible] R
	RE-02-B	Reactor Water Level Switch	RKO-2	72'	Cleanup System	215°	Fig. 11	16 PSIA	<6.1 x 10[illegible] R
	RE-02-C	Reactor Water Level Switch	RKO-1	72'	Emer. Cond.	230°	Fig. 4	16 PSIA	1.4 x 10[illegible] R
	RE-02-D	Reactor Water Level Switch	RKO-2	72'	Cleanup System	215°	Fig. 11	16 PSIA	<6.1 x 10[illegible] R
27	V-23-1*	Purge Valves	Top of Torus x NE of Reactor Vessel	19'	Main Steam	160°*	Fig. 15*	16 PSIA*	<6.1 x 10[illegible] R
	V-23-2*	Purge Valves	Top of Torus x NE of Reactor Vessel	19'	Main Steam	160°*	Fig. 15*	16 PSIA*	<6.1 x 10[illegible] R
	V-23-3*	Purge Valves	℄ Reactor Vessel West x $R_E - R_F$	63'	Emer. Cond.	250°*	Fig. 9*	16 PSIA*	1.3 x 10[illegible] R
	V-23-4*	Purge Valves	℄ Reactor Vessel West x $R_E - R_F$	63'	Emer. Cond.	250°*	Fig. 9*	16 PSIA*	1.3 x 10[illegible] R
	V-23-13*	Nitrogen Valves	℄ Reactor Vessel West x $R_E - R_F$	63'	Emer. Cond.	250°*	Fig. 9*	16 PSIA*	1.3 x 10[illegible] R
	V-23-14*	Nitrogen Valves	℄ Reactor Vessel West x $R_E - R_F$	63'	Emer. Cond.	250°*	Fig. 9*	16 PSIA*	1.3 x 10[illegible] R
	V-23-17*	Nitrogen Valves	℄ Reactor Vessel West x $R_E - R_F$	63'	Emer. Cond.	250°*	Fig. 9*	16 PSIA*	1.3 x 10[illegible] R
	V-23-18*	Nitrogen Valves	℄ Reactor Vessel West x $R_E - R_F$	63'	Emer. Cond.	250°*	Fig. 9*	16 PSIA*	1.3 x 10[illegible] R

FIGURE SUPPLIED
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POOR ORIGINAL

Table A-1. Oyster Creek Nuclear Generating Station
Electrical Equipment Environmental Conditions (Cont.)

*Note: the listed equipment is not required to mitigate the consequences of the accident outside of containment or to achieve a safe shutdown for that accident. For a break inside containment, asterisked items are needed to mitigate the accident, however, the environmental conditions for these asterisked items would be normal ambient conditions.

Target Designation	Description	Location Coordinates	Approximate Elevation	Worst Case Line Breaks	Peak Temperature (°F)	Temperature Profile	Peak Pressure PSIA	Total Integrated Radiation Exposure (1 Yr) NUREG 0578 Assumptions (RA)
28								
V-23-15*	Nitrogen System Valves	R_2 R_3 x R_E R_F (by Wall)	33'	-	170°*	--	16PSIA*	$2.9x10^5$ R
V-23-16*	Nitrogen System Valves	R_2 R_3 x R_E R_F (by Wall)	33'	-	170°*	--	16PSIA*	$2.9x10^5$ R
V-23-19*	Nitrogen System Valves	R_2 R_3 x R_E R_F (by Wall)	33'	-	170°*	--	16PSIA*	$2.9x10^5$ R
V-23-20*	Nitrogen System Valves	R_2 R_3 x R_E R_F (by Wall)	33'	-	170°*	--	16PSIA*	$2.9x10^5$ R
29 V-38-16*	Particulate Monitoring System	R_C x NW Drywell Wall	27'	-	170°*	--	16PSIA*	$7.6x10^4$ R
V-38-17*	Particulate Monitoring System	R_C x NW Drywell Wall	27'	-	170°*	--	16PSIA*	$7.6x10^4$ R
V-38-9*	Oxygen Analyzer System	R_C x NW Drywell Wall	27'	-	170°*	--	16PSIA*	$7.6x10^4$ R
V-38-10*	Oxygen Analyzer System	R_C x NW Drywell Wall	27'	-	170°*	--	16PSIA*	$7.6x10^4$ R
V-38-22*	Torus Sample System	R_C x NW Drywell Wall	27'	-	170°*	--	16PSIA*	$7.6x10^4$ R
V-38-23*	Torus Sample System	R_C x NW Drywell Wall	27'	-	170°*	--	16PSIA*	$7.6x10^4$ R
30 V-28-21*	Ventilation Valve	Top of Torus x NE Reactor Vessel Wall	19'	Main Steam	150°*	Fig. 15*	16PSIA*	<6.1x10⁴ R
V-28-22*	Ventilation Valve	Top of Torus x NE Reactor Vessel Wall	19'	Main Steam	150°*	Fig. 15*	16PSIA*	$<6.1x10^4$ R
V-28-17*	Ventilation Valve	Top of Torus x Reactor Vessel East Wall	19'	Main Steam	150°*	Fig. 15*	16PSIA*	$<6.1x10^4$ R
V-28-18*	Ventilation Valve	Top of Torus x Reactor Vessel East Wall	19'	Main Steam	150°*	Fig. 15*	16PSIA*	$<6.1x10^4$ R
V-28-47*	Ventilation Valve	Top of Torus x Reactor Vessel East Wall		Main Steam	150°*	Fig. 15*	16PSIA*	$<6.1x10^4$ R
31								
IB-18 ([illegible] switches)	Temperature Switch for Reactor Isolation	¶. Steam Tunnel xR_G R_F	32'3" (6' above steam line)	Main Steam	280°	Fig. 3	21PSIA	$<6.1x10^4$ R
IB-18 ([illegible] switches)	Temperature Switch for Reactor Isolation	¶. Steam Tunnel xR_H R_J	32'3" (6' above steam line)	Main Steam	290°	Fig. 1	28 PSIA	$<6.1x10^4$ R
IB-18 ([illegible] switches)	Temperature Switch for Reactor Isolation	¶. Steam Tunnel x RH	32'3" (6' above steam line)	Main Steam	293°	Fig. 1	28 PSIA	$<6.1x10^4$ R
IB-19 ([illegible] switches)	Temperature Switch for Reactor Isolation	¶. Steam Tunnel x R_F xR_H	32'3" (6' above steam line)	Main Steam	290°	Fig. 1	28 PSIA	$<6.1x10^4$ R
IB-19 ([illegible] switches)	Temperature Switch for Reactor Isolation	¶. Steam Tunnel x R_F	32'3" (6' above steam line)	Main Steam	290°	Fig. 1	28 PSIA	$<6.1x10^4$ R

FIGURE SUPPLIED BY THE LICENSEE

POOR ORIGINAL

Table A-1. Oyster Creek Nuclear Generating Station
Electrical Equipment Environmental Conditions (Cont.)

*Indicates that equipment is not required to mitigate the consequences of the accident outside of containment or to achieve a safe shutdown for that accident. For a break inside containment, asterisked items are needed to mitigate the accident, however, the environmental conditions for these asterisked items would be normal ambient conditions.

	Target Designation	Description	Location Coordinates	Approximate Elevation	Worst Case Line Break	Peak Temperature (°F)	Temperature Profile	Peak Pressure PSIA	Total Integrated Radiation Exposure (1 Yr. NUREG 0578 Assumptions (R)
12.	NZ 01 A	Core Spray Pump	Northwest Cor. Rm.	( ) 16' 0"	-	77°	-	15 PSIA	6.6×10^5R
	NZ 01 B	Core Spray Pump	Southwest Cor. Rm.	( ) 16' 0"	-	77°	-	16 PSIA	6.6×10^5R
	NZ 01 C	Core Spray Pump	Northwest Cor. Rm.	( ) 16' 0"	-	77°	-	15 PSIA	6.6×10^5R
	NZ 01 D	Core Spray Pump	Southwest Cor. Rm.	( ) 16' 0"	-	77°	-	15 PSIA	6.6×10^5R
13.	PM 51 1 1	Contnd. Spray Pump	Northeast Cor. Rm.	( ) 18' 0"	Main Steam	165°	Fig. 18	15 PSIA	6.1×10^5R
	PM 51 1 2	Contnd. Spray Pump	Northeast Cor. Rm.	( ) 18' 0"	Main Steam	165°	Fig. 18	15 PSIA	6.3×10^5R
	PM 51 1 3	Contnd. Spray Pump	Southeast Cor. Rm.	( ) 18' 0"	Main Steam	165°	Fig. 14	15 PSIA	1.0×10^6R
	PM 51 1 4	Contnd. Spray Pump	Southeast Cor. Rm.	( ) 18' 0	Main Steam	166°	Fig. 14	16 PSIA	1.0×10^6R
14.	IP 05A*	Contnd. Spray Diff. Press. Trans.	$R_6\ R_7 \times R_A\ R_B$	29' 0"	Cleanup System	140°*	Fig. 13*	15 PSIA*	7.5×10^5R
	IP 05B*	Contnd. Spray Diff. Press. Trans.	$R_6\ R_7 \times R_A\ R_B$	29' 0"	Cleanup System	140°*	Fig. 13*	15 PSIA*	7.5×10^5R
	IP 05C*	Contnd. Spray Diff. Press. Trans.	$R_1\ R_2 \times R_B\ R_C$	29' 0"	-	77°*	-	15 PSIA*	7.5×10^5R
	IP 05D*	Contnd. Spray Diff. Press. Trans.	$R_1\ R_2 \times R_B\ R_C$	29' 0"	-	77°*	-	15 PSIA*	7.5×10^5R
15.	NS 01A 11	MSIV Solenoid Valves	Steam Tunnel	30' 6"	Main Steam	280°	Fig. 3	21 PSIA	$<6.1 \times 10^4$R
	NS 01A 12	MSIV Solenoid Valves	Steam Tunnel	30' 6"	Main Steam	280°	Fig. 3	21 PSIA	$<6.1 \times 10^4$R
	NS 01A 13	MSIV Solenoid Valves	Steam Tunnel	30' 6"	Main Steam	280°	Fig. 3	21 PSIA	$<6.1 \times 10^4$R
	NS 01B 11	MSIV Solenoid Valves	Steam Tunnel	30' 6"	Main Steam	280°	Fig. 3	21 PSIA	$<6.1 \times 10^4$R
	NS 01B 12	MSIV Solenoid Valves	Steam Tunnel	30' 6"	Main Steam	280°	Fig. 3	21 PSIA	$<6.1 \times 10^4$R
	NS 01B 13	MSIV Solenoid Valves	Steam Tunnel	30' 6"	Main Steam	280°	Fig. 3	21 PSIA	$<6.1 \times 10^4$R
	NS 01A 1	MSIV Position Indicators	Steam Tunnel	30' 6"	Main Steam	280°	Fig. 3	21 PSIA	$<6.1 \times 10^4$R
	NS 01A 2	MSIV Position Indicators	Steam Tunnel	30' 6"	Main Steam	280°	Fig. 3	21 PSIA	$<6.1 \times 10^4$R
	NS 01B 1	MSIV Position Indicators	Steam Tunnel	30' 6"	Main Steam	280°	Fig. 3	21 PSIA	$<6.1 \times 10^4$R
	NS 01B 2	MSIV Position Indicators	Steam Tunnel	30' 6"	Main Steam	280°	Fig. 3	21 PSIA	$<6.1 \times 10^4$R
16.	V 16 2	Clean Up Valve	$R_1\ R_2 \times R_D\ R_E$	69' 6"	Cleanup System	145°	Fig. 16	15 PSIA	1.5×10^4R
	V 16 11	Clean Up Valve	$R_1\ R_2 \times R_D\ R_E$	69' 6"	Cleanup System	145°	Fig. 16	15 PSIA	1.5×10^4R
	V 16 61	Clean Up Valve	$R_1\ R_2 \times R_D\ R_E$	68' 10"	Cleanup System	145°	Fig. 16	15 PSIA	1.5×10^4R
17.	V 21 30*	Reactor Water Sample Valve	$R_2\ R_3 \times R_D\ R_E$	55' 0"	Cleanup System	215°*	Fig. 17*	15 PSIA*	4.7×10^4R
18.	V 17 1*	Shutdown Cooling Valve	$R_5\ R_6 \times R_B\ R_C$	46' 2"	-	77°*	-	16 PSIA*	1.4×10^6R
	V 17 2*	Shutdown Cooling Valve	$R_3\ R_4 \times R_B\ R_C$	46' 2"	-	77°*	-	15 PSIA*	1.4×10^6R
	V 17 3*	Shutdown Cooling Valve	$R_6\ R_7 \times R_B\ R_C$	46' 2"	-	77°*	-	16 PSIA*	1.4×10^5R
	V 17 55*	Shutdown Cooling Valve	$R_5\ R_6 \times R_C\ R_D$	51' 9"	-	77°*	-	16 PSIA*	8.5×10^5R
	V 17 56*	Shutdown Cooling Valve	$R_5\ R_6 \times R_C\ R_D$	51' 9"	-	77°*	-	15 PSIA*	1.1×10^6R
	V 17 57*	Shutdown Cooling Valve	$R_5\ R_6 \times R_C\ R_D$	51' 9"	-	77°*	-	15 PSIA*	1.1×10^6R

FIGURE SUPPLIED BY THE LICENSEE

POOR ORIGINAL

Table A-1. Oyster Creek Nuclear Generating Station
Electrical Equipment Environmental Conditions (Cont.)

[illegible] outside of containment or to achieve a safe shutdown for that accident. For a break inside containment, asterisked items are needed to mitigate the accident, however, the environmental conditions for these asterisked items would be normal ambient conditions.

	Target Designation	Description	Location Coordinates	Approximate Elevations	Worst Case Line Break	Peak Temperature (°F)	Temperature Profiles	Peak Pressure (PSIA)	Total Integrated Radiation Exposure (1 Yr NUREG 0578 Assumptions) (RA)
39.	V 22 1*	Drywell Sump Disch. Valve	$R_5\ R_6 \times R_B\ R_C$	39' 9"	-	77°*	-	15PSIA*	6.4×10^5R
	V 22 2*	Drywell Sump Disch. Valve	$R_5\ R_6 \times R_B\ R_C$	39' 9"	-	77°*	-	15PSIA*	6.4×10^5R
	V 22 28*	Drywell Sump Disch. Valve	$R_5\ R_6 \times R_B\ R_C$	39' 9"	-	77°*	-	15PSIA*	6.4×10^5R
	V 22 29*	Drywell Sump Disch. Valve	$R_5\ R_6 \times R_B\ R_C$	39' 9"	-	77°*	-	15PSIA*	6.4×10^5R
40.	V 20 15	Core Spray Valve	$R_5\ R_6 \times R_D\ R_E$	65' 6"	Emer. Cond.	230°	Fig. 4	16PSIA	3.9×10^5R
	V 20 10	Core Spray Valve	$R_5\ R_6 \times R_D\ R_E$	58' 10"	Emer. Cond.	230°	Fig. 4	16PSIA	3.9×10^5R
41.	IS 1a*	Condnd. Iso. Valve Switch	$R_6\ R_7 \times R_D\ R_E$	65' 0"	Emer. Cond.	230°*	Fig. 4*	16PSIA*	1.5×10^6R
42.	NZ 03 A	Core Spray Boost Pump	$R_6\ R_7 \times R_D\ R_E$	51' 1"	Emer. Cond.	230°	Fig. 4	16PSIA	1.2×10^6R
	NZ 03 B	Core Spray Boost Pump	$R_2\ R_3 \times R_E$	26' 1"	-	77°	-	15PSIA	2.9×10^5R
	NZ 03 C	Core Spray Boost Pump	$R_6\ R_7 \times R_D\ R_E$	51' 1"	Emer. Cond.	230°	Fig. 4	16PSIA	1.2×10^6R
	NZ 03 D	Core Spray Boost Pump	$R_1\ R_2 \times R_E$	26' 4"	-	77°	-	15PSIA	2.9×10^5R
43.	V 20 21	Core Spray Valve	$R_2\ R_3 \times R_D\ R_E$	83' 4"	Emer. Cond.	225°	Fig. 8	16PSIA	8.8×10^5R
	V 20 41	Core Spray Valve	$R_2\ R_3 \times R_D\ R_E$	83' 1"	Emer. Cond.	225°	Fig. 8	16PSIA	8.8×10^5R
44.	V 14 30	Emerg. Cond. Valve	$R_3\ R_4 \times R_B\ R_C$	90' 0"	Emer. Cond.	280°	Fig. 6	16PSIA	1.0×10^6R
	V 14 31	Emerg. Cond. Valve	$R_3\ R_4 \times R_B\ R_C$	90' 0"	Emer. Cond.	280°	Fig. 6	16PSIA	1.0×10^6R
	V 14 32	Emerg. Cond. Valve	$R_4\ R_5 \times R_B\ R_C$	90' 0"	Emer. Cond.	280°	Fig. 6	16PSIA	9.4×10^5R
	V 14 33	Emerg. Cond. Valve	$R_4\ R_5 \times R_B\ R_C$	90' 0"	Emer. Cond.	280°	Fig. 6	16PSIA	9.4×10^5R
	V 14 34	Emerg. Cond. Valve	$R_4\ R_5 \times R_B\ R_C$	86' 10"	Emer. Cond.	280°	Fig. 6	16PSIA	9.4×10^5R
	V 14 35	Emerg. Cond. Valve	$R_3\ R_4 \times R_B\ R_C$	87' 5"	Emer. Cond.	280°	Fig. 6	16PSIA	1.0×10^6R
45.	V 14 34	Emerg. Cond. Valve	$R_4\ R_5 \times R_A\ R_B$	99' 9"	Emer. Cond.	270°	Fig. 5	16PSIA	5.3×10^5R
	V 14 36	Emerg. Cond. Valve	$R_4\ R_5 \times R_B\ R_C$	99' 9"	Emer. Cond.	270°	Fig. 5	16PSIA	5.3×10^5R
46.	RE 18 A	Reactor Vessel Level Switch	RKO 1	72'	Emer. Cond.	230°	Fig. 4	16PSIA	1.4×10^4R
	RE 13 B	Reactor Vessel Level Switch	RKO 2	72'	Cleanup System	215°	Fig. 11	16PSIA	$<6.1 \times 10^4$R
	RE 13 C	Reactor Vessel Level Switch	RKO 1	72'	Emer. Cond.	230°	Fig. 4	16PSIA	1.4×10^4R
	RE 13 D	Reactor Vessel Level Switch	RKO 2	72'	Cleanup System	215°	Fig. 11	16PSIA	$<6.1 \times 10^4$R
	RE 05 19 A	Reactor Vessel Level Switch/Trans.	RKO 1	72'	Emer. Cond.	230°	Fig. 4	16PSIA	1.4×10^4R
	RE 05 19 B	Reactor Vessel Level Switch/Trans.	RKO 2	72'	Cleanup System	215°	Fig. 11	16PSIA	$<6.1 \times 10^4$R

FIGURE SUPPLIED BY THE LICENSEE

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APPENDIX B - LISTING OF SAFETY-RELATED ELECTRICAL EQUIPMENT

The following table lists the groupings of safety-related electrical equipment items for the Oyster Creek Nuclear Generating Station. Equipment item numbers provided in the table are used in the Equipment Environmental Qualification Summary Forms and in the equipment qualification discussions presented in Section 4.

This table was generated from the lists of equipment items provided by the Licensee in Reference 1. FRC has listed plant equipment items by manufacturer and model number, plant location, and time required to function as identified by the Licensee.

DELETED MATERIAL IS PROPRIETARY INFORMATION

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	SCEWS NO.	QUALIFICATION REFERENCES	TIME REQUIRED
1	Pressure Switches Dresser 1593 VX (Automatic Depressurization)	Reactor Building	1-5	None	Intermediate (~ 3 h)
2	Solenoid Valves ASCO NP-8344A70E	Reactor Building	6, 7	2.24	Long (~ 30 days)
3A	Motorized Valve Actuators Limitorque SMB-00 (Containment Spray)	Reactor Building	8, 9	2.2, 2.3, 2.4, 2.5	Long (30 days)
3B	Motorized Valve Actuators Limitorque SMB-00 (Containment Spray)	Reactor Building	14-17	2.2, 2.3, 2.4, 2.5	Intermediate (4 h)
4A	Motorized Valve Actuators Limitorque SMB-000 (Drywell Isolation)	Reactor Building	10, 11	2.2, 2.3, 2.4, 2.5	Long (30 days)
4B	Motorized Valve Actuators Limitorque SMB-000 (Containment Spray)	Reactor Building	12, 13	2.2, 2.3, 2.4, 2.5	Intermediate (4 h)
4C	Motorized Valve Actuators Limitorque SMB-000 (MSIV)	Drywell	I-2A	None	Short (2 min.)
5	Pressure Switch Static-O-Ring 12NKA (Drywell Pressure Scram)	Reactor Building	18-21	2.6, 2.7	Long (30 days)

DELETED MATERIAL IS PROPRIETARY INFORMATION

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	SCEWS NO.	QUALIFICATION REFERENCES	TIME REQUIRED
6	Pressure Trans- mitters GE/MAC 551 (Reactor Vessel Pressure)	Reactor Building	22, 23	None	Long (30 days)
7	Pressure Trans- mitters GE/MAC 551 (Reactor Vessel Pressure)	Reactor Building	24, 25	None	Long (30 days)
8A	Level Transmitters General Electric GE/MAC 553 (Isolation Condenser Level)	Reactor Building	26 29	None	Intermediate (4 h)
8B	Level Transmitters General Electric GE/MAC 553 (Reactor Water Level)	Reactor Building	30-33	None	Long (30 days)
8C	Flow Transmitter General Electric GE/MAC 553 (Containment Spray Flow)	Reactor Building	48,49	2.25	Short (4 h)
8D	Pressure Transmitter General Electric GE/MAC 553 (Drywell Pressure)	Reactor Building	54	2.25	Long (30 days)
8E	Pressure Transmitter General Electric GE/MAC 553 (Containment Spray Differential)	Reactor Building	133-136	2.25	Long (30 days)
9	Pressure Switches Mercoid 9-51/DAW- 43-156-R2IE (Core Spray)	Reactor Building	34-37, 39,41	None	Long (30 days)

DELETED MATERIAL IS PROPRIETARY INFORMATION

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	SCEWS NO.	QUALIFICATION REFERENCES	TIME REQUIRED
10A	Pressure Switches General Electric GE/MAC 552 (Core Spray)	Reactor Building	42, 43	None	Long (30 days)
10B	Pressure Switches Mercoid 9-51/DAW-43-156-R21E (Core Spray)	Reactor Building	38, 40	None	Long (30 days)
11	Temperature Detectors Rochester Instruments No Model No. (Isolation Condenser Area Leak Detection)	Reactor Building	44-47	None	Short (10 min.)
12A	Pressure Switches Barton 288A (Containment Pressure)	Reactor Building	50-53, 55-58	2.7, 2.10	Long (30 days)
12B	Reactor Isolation Switches Barton 288A (Reactor Isolation)	Reactor Building	63-70	2.7, 2.10	Long (30 days)
12C	Level Switches Barton 288A (Reactor Vessel Level)	Reactor Building	178-181	None	Long (30 days)
12D	Level Switches Barton 288A (Isolation Condenser Delta P)	Reactor Building	71-78	None	Intermediate (4 h)
13	Pressure Switches Meletron 372 (MSL Low Pressure)	Reactor Building	59-62	2.7	Long (30 days)

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	SCEWS NO.	QUALIFICATION REFERENCES	TIME REQUIRED
14A	Pressure Switches Barksdale B2T-A12SS (Core Spray)	Reactor Building	79, 80	None	Long (30 days)
14B	Pressure Switches Barksdale B2T-A12SS (Reactor Pressure)	Reactor Building	87-90	None	Short (10 min.)
15	Pressure Switches Barksdale E2T-M12SS (Core Spray)	Reactor Building	81, 82	None	Long (30 days)
16	Pressure Switches Barksdale B2T (Reactor Vessel Pressure)	Reactor Building	83-86	None	Short (3 h)
17	Level Switches Yarway 4316E (Reactor Water Level)	Reactor Building	91, 92	2.11, 2.12	Short (10 min.)
18	Level Switches Yarway C2337 (Reactor Water Level)	Reactor Building	93-96, 182, 183	2.11, 2.12	Long (30 days)
19	Solenoid Valves ASCO 8344-B27 (Drywell Isolation)	Reactor Building	97, 98	2.7, 2.13	Long (30 days)
20	Solenoid Valves ASCO 8344-A27 (Drywell Isolation)	Reactor Building	99, 100, 102	2.7, 2.11	Long (30 days)
21A	Solenoid Valves ASCO 83148 (Drywell Isolation)	Reactor Building	101	2.7, 2.11	Long (30 days)

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	SCEWS NO.	QUALIFICATION REFERENCES	TIME REQUIRED
21B	Solenoid Valves ASCO 83148 (Isolation Condenser)	Reactor Building	176, 177	2.11	Intermediate (4 h)
22A	Solenoid Valves ASCO WP8300B61RU (Drywell Isolation)	Reactor Building	103, 104,	2.7,2.11	Long (30 days)
22B	Solenoid Valves ASCO WP8300B61RU (Drywell Isolation)	Reactor Building	115-119	2.7,2.11	Long (30 days)
23	Solenoid Valve ASCO WPLB83177 (Drywell Isolation)	Reactor Building	105	2.6,2.7	Long (30 days)
24	Solenoid Valve ASCO 831424 (Drywell Isolation)	Reactor Building	106	2.6,2.7	Long (30 days)
25	Solenoid Valve ASCO X8031A42 (Drywell Isolation)	Reactor Building	107	2.6,2.7	Long (30 days)
26	Solenoid Valves Atkomatic 15-702-B (50R) (Drywell Isolation)	Reactor Building	109-112	2.6,2.7	Long (30 days)
27	Solenoid Valves ASCO LB82627 (Drywell Isolation)	Reactor Building	113, 114	2.7	Long (30 days)
28	Temperature Switches Fenwal 17002-40 (MSL Leak Detection)	Main Steam Tunnel	120-124	2.11	Short (1 min.)

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	SCEWS NO.	QUALIFICATION REFERENCES	TIME REQUIRED
29	Electric Motors General Electric 5K-828848C7 (Core Spray Pumps)	Reactor Building	125-128	2.14	Long (30 days)
30	Electric Motors General Electric 5K-818842A103 (Containment Spray Pumps)	Reactor Building	129-132	2.14	Intermediate (4 h)
31A	Solenoid Valves ASCO 206-832-3RU (MSIV)	Reactor Building	137-139	2.24	Short (1 min.)
31B	Solenoid Valves ASCO 206-832-3RU (MSIV)	Drywell	I-1A	2.16, 2.24	Short (2 min.)
32A	Solenoid Valves ASCO 206-301-3R (MSIV)	Reactor Building	140-141	2.24	Short (1 min.)
32B	Solenoid Valves ASCO 206-301-3RU (MSIV)	Drywell	I-1B	2.16, 2.24	Short (2 min.)
33	Position Switches Snaplock (NAMCO) SL3-C58W (MSIV Position Indication)	Reactor Building	143-146	2.11	Short (1 min.)
34A	Motorized Valve Actuators Limitorque SMB-0 (Drywell Isolation)	Reactor Building	147-149	2.2, 2.3, 2.4, 2.5	Short (1 min.)

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	SCEWS NO.	QUALIFICATION REFERENCES	TIME REQUIRED
34B	Motorized Valve Actuators Limitorque SMB-0 (Drywell Isolation)	Reactor Building	151-156	2.2, 2.3, 2.4, 2.5	Long (30 days)
34C	Motorized Valve Actuators Limitorque SMB-0 (Shutdown Cooling)	Drywell	I-2B	2.16	Short (2 min.)
35	Solenoid Valve ASCO LM831424 (Drywell Isolation)	Reactor Building	150	2.7, 2.11	Long (30 days)
36	Solenoid Valves ASCO WP8300B61U (Drywell Isolation)	Reactor Building	157-160	2.7	Long (30 days)
37	Motorized Valve Actuators Limitorque SMB-1 (Core Spray)	Reactor Building	161, 162, 168, 169	2.2,2.3, 2.4,2.5	Long (30 days)
38	Isolation Valve Switch Meletron 4201E-3B (Drywell Pressure)	Reactor Building	163	2.7	Long (30 days)
39	Electric Motors General Electric 5K-818841C45 (Core Spray Booster Pump)	Reactor Building	164-167	2.14	Long (30 days)
40	Motorized Valve Actuators Limitorque SMB-2 (Isolation Condenser)	Reactor Building	170-175	2.2, 2.3, 2.4, 2.5	Long (30 days)

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	SCEWS NO.	QUALIFICATION REFERENCES	TIME REQUIRED
41	Level Switch Magnetrol S1M3 (Scram Discharge Volume Level)	Reactor Building	184-187	None	Short (5 min.)
42	Solenoid Valve ASCO WT8300B61RV (Drywell Isolation)	Reactor Building	188	None	Short (5 min.)
43	Solenoid Valve ASCO NP-8320A187E (Sample Valve)	Drywell	I-1C	2.16, 2.24	Short (2 min.)
44	Motorized Valve Actuator Limitorque SMB-2 (Isolation Condenser)	Drywell	I-2D	2.16	Short (2 min.)
45	Electrical ' tration General Elec.,ic F01	Drywell	I-3	2.17, 2.18, 2.19	Long (30 days)
46	Electrical Connectors ITT-Cannon CA-3106E-36A-46P-F80 CA-3100K-36A-46S-F80	Drywell	I-4B	2.7, 2.16	Long (30 days)
47	Electrical Connectors ITT-Cannon CA 06RX-36A-10P-A95 CA 3100RX-36A-10S-A95	Drywell	I-4D	2.7, 2.16	Long (30 days)
48	Terminal Blocks General Electric EB	Drywell	I-5	2.16, 2.20	Long (30 days)
49	Electrical Cable General Electric Vulkene SI-58145	Drywell	I-6A	2.16, 2.21	Long (30 days)

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	SCEWS NO.	QUALIFICATION REFERENCES	TIME REQUIRED
50	Electrical Cable General Electric S1-58073	Drywell	I-6A	2.16, 2.21	Long (30 days)
51	Electrical Cable Tensolite Co. No Model No.	Drywell	I-6B	2.16, 2.22	Long (30 days)
52	Electrical Cable Kerite No Model No.	Drywell	I-6C	2.16, 2.23	Long (30 days)
53	Electrical Cable Rockbestos	Drywell	I-7A, I-7B	2.15, 2.16	Long (30 days)
54	Electrical Cable Splices Raychem Corp. WCSF	Drywell	I-8	2.9, 2.16	Long (30 days)
55	Solenoid Valve Dresser 1525 VX (PORV)	Drywell	I-9	2.1, 2.16	Intermediate (4 h)
56	Solenoid Valve ASCO No Model No. (Drywell Isolation)	Reactor Building	108	None	Long (30 days)

APPENDIX C - SAFETY SYSTEMS AND DISPLAY INSTRUMENTATION FOR WHICH ENVIRONMENTAL QUALIFICATION IS TO BE ADDRESSED

The NRC transmitted to the Licensees for the SEP plants, Indian Point Units 2 and 3, and Zion Units 1 and 2 the DOR Guidelines for evaluating Class 1E equipment qualification and the "Guidelines for Identification of That Safety Equipment of SEP Operating Reactors for Which Environmental Qualification is To Be Addressed." Based on these documents, the Licensee submitted a list of safety-related systems that must function in order to mitigate the consequences of a design basis accident. As a result of discussions between the Licensee and the NRC, the following list represents systems and display instruments for which the Licensee and the NRC have determined that qualification is to be addressed.

A. Safe Shutdown Systems

Reactor Protection System*
Isolation Condenser*
Demineralized Water Transfer+
Service Water
Radiation Monitoring++
Sampling++
Emergency Diesel AC Power+
125 V DC Power*
Emergency Power Distribution*

B. Accident Mitigating Systems (LOCA, MSLB, FWLB)

Safeguards Actuation System
Reactor Depressurization System
Core Spray
Main Steam Isolation
Containment Isolation
Containment Spray
Standby Gas Treatment+
Combustible Gas Control

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*Systems used for both safe shutdown and accident mitigation.
+Review of this equipment deferred until after February 1, 1981, as referenced in Section 2.2.3.
++To be added as TMI-Lessons Learned requirement.

C. Accident Mitigating and Safe Shutdown Instruments (LOCA, MSLB, FWLB)

Reactor Water Level
Reactor Steam Pressure
Containment Drywell Pressure**
Containment Torus Water Level**
Containment Spray Flow**
Isolation Condenser Shell-Side Water Level
Emergency Service Water Pump Discharge Pressure
Containment Spray Pump Suction Pressure**
Demineralized Water Pump Discharge Pressure

**Instruments needed for accident mitigation purposes only.

APPENDIX D - EVALUATION OF LICENSEE JUSTIFICATIONS FOR CONTINUED OPERATION

The Licensee's documentation contained justification for interim plant operation where qualification had not been demonstrated for certain equipment items. At the request of the NRC, FRC conducted a technical evaluation of these justifications based upon a review of technical information made available by the Licensee.

In Chapter 7 of the Licensee's final submittal [1], the Licensee presents "justifications for continued operation" for equipment items that presently lack complete qualification.

FRC has performed technical evaluations of each of the positions which the Licensee presents in Chapter 7. FRC finds no technical deficiencies in these positions with the exception of five minor concerns that are expressed below. FRC's concerns involve the Licensee's assertions that these equipment items are not required to mitigate the consequences of an HELB or used for safe shutdown of the plant.

Paragraph 19, Chapter 7. The Licensee has indicated that the main steam line low pressure switches are exposed to a peak temperature and pressure resulting from a feedwater line break and are protected from the consequences of a MSLB. The actual temperature and pressure resulting from a MSLB have not been identified. FRC believes that these conditions should be established and that environmental qualification should be addressed for the main steam line low pressure switches (Equipment Item No. 13).

Paragraph 27, Chapter 7. It is not clear that the need to periodically purge the containment throughout the long-term cooling period following an HELB outside containment is totally unnecessary such that the purge valves (Equipment Item Nos. 19, 20, 21A, and 22A) may be allowed to become inoperative. FRC believes that these valves should be qualified for their post-accident environment.

Paragraph 28, Chapter 7. Similarly, it is not clear that there is no need to open the nitrogen system purge valves at some time following a LOCA. Therefore, the solenoid valves controlling these pneumatic valves (Equipment Items Nos. 23, 24, 25, and 56) should be qualified to operate under the environmental service conditions to which they may be subjected.

Paragraph 37, Chapter 7. If sample valve V-24-30 (Equipment Item No. 35) is open coincident with an HELB, the inside containment isolation valve cannot be relied upon to perform its isolation function in view of the single active failure criterion. FRC believes valve V-24-30 should be qualified for the post-accident environment to which it is exposed. Furthermore, the qualification of this valve is required in order to obtain post-accident samples in accordance with the recommendations of the TMI-2 Lessons Learned Task Force.

Paragraph 39, Chapter 7. FRC believes that drywell sump discharge valves (Equipment Item No. 36) should be qualified for their post-accident environment for long-term service because they will eventually need to be opened in order to remove contained fluids.

DELETED MATERIAL IS PROPRIETARY INFORMATION

APPENDIX E - CORRELATION OF EQUIPMENT ITEM NUMBERS WITH REPORT SECTIONS OF DRAFT INTERIM AND FINAL TECHNICAL EVALUATION REPORTS

EQUIPMENT ITEM NO.	DRAFT INTERIM TECHNICAL EVALUATION REPORT SECTION	FINAL TECHNICAL EVALUATION REPORT SECTION
1	None	4.5.2.1
2	None	4.3.1.1
3A	None	4.3.3.1
3B	None	4.3.3.1
4A	None	4.3.1.5
4B	None	4.3.3.1
4C	3.3.2.1	4.5.2.2
5	None	4.7.1
6	None	4.6.1
7	None	4.6.2
8A	None	4.6.3
8B	None	4.6.4
8C	None	4.6.3
8D	None	4.6.4
8E	None	4.6.4
9	None	4.7.2
10	None	4.6.14
11	None	4.5.2.3
12A	None	4.7.3
12B	None	4.7.4
12C	None	4.6.15
12D	None	4.6.10
13	None	4.7.11
14A	None	4.6.16
14B	None	4.7.5
15	None	4.6.16
16	None	4.7.6
17	None	4.7.7
18	None	4.6.17
19	None	4.5.2.4
20	None	4.5.2.4
21A	None	4.5.2.4
21B	None	4.6.13
22A	None	4.5.2.4
22B	None	4.5.2.5
23	None	4.6.9
24	None	4.6.9
25	None	4.6.9
26	None	4.5.2.6

CORRELATION OF EQUIPMENT ITEM NUMBERS WITH
REPORT SECTIONS OF DRAFT INTERIM
AND FINAL TECHNICAL EVALUATION REPORTS (Cont.)

EQUIPMENT ITEM NO.	DRAFT INTERIM TECHNICAL EVALUATION REPORT SECTION	FINAL TECHNICAL EVALUATION REPORT SECTION
27	None	4.5.2.7
28	None	4.5.2.8
29	None	4.7.10
30	None	4.6.5
31A	None	4.3.1.4
31B	3.2.1	4.5.2.9
32A	None	4.3.1.4
32B	3.2.1	4.5.2.9
33	None	4.6.11
34A	None	4.3.1.5
34B	None	4.3.3.1
34C	3.3.2.1	4.5.2.10
35	None	4.5.2.11
36	None	4.5.2.11
37	None	4.5.2.12
38	None	4.7.8
39	None	4.4.1
40	None	4.5.2.13
41	None	4.7.9
42	None	4.5.2.14
43	3.2.1	4.5.2.15
44	3.3.2.1	4.5.2.10
45	3.3.2.2	4.6.6
46	3.3.2.3	4.5.2.16
47	3.3.2.3	4.5.2.16
48	3.3.2.4	4.6.7
49	3.3.2.5	4.3.1.2
50	3.3.2.5	4.3.1.2
51	3.3.4.1	4.5.2.17
52	3.3.2.6	4.3.3.2
53	3.2.2	4.3.1.3
54	3.3.2.7	4.3.3.3
55	3.3.2.8	4.6.12
56	None	4.6.8

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APPENDIX F - PROPERTIES OF CAST PHENOLIC RESINS

PHYSICAL PROPERTIES

	Specific Gravity	Specific Heat	Thermal Conductivity (c.g.s. units) $\times 10^{-4}$	Thermal Expansion Coefft. (per °C) $\times 10^{-5}$	Water Absorption* (mg)
Cast Resin	1.25-1.32	0.4-0.5	3-5	3-9	2-20
Moulding Material					
Wood-flour-filled	1.3-1.4	0.35-0.36	4-12	3-6	70-150
Chopped-cotton-fabric-filled	1.3-1.4	0.30-0.35	3-5	2-6	200-400
Mineral-filled	1.6-2.4	0.25-0.35	8-20	2-4	20-100
Laminated Material					
Paper-filled	1.3-1.4	0.3-0.4	3-8	2-3	15-300
Fabric-filled	1.3-1.4	0.3-0.4	5-8	2-3	200-300
Asbestos-filled	1.5-2.0	0.25-0.35	8-20	2-3	100-200

MECHANICAL PROPERTIES

	Ultimate Tensile Strength (lbf/in²) $\times 10^3$	Bending Strength (lbf/in²) $\times 10^3$	Ultimate Shear Strength (lbf/in²) $\times 10^3$	Ultimate Compression Strength (lbf/in²) $\times 10^3$	Modulus of Elasticity (in tension) (lbf/in²) $\times 10^3$	Modulus of Rigidity (in torsion) (lbf/in²) $\times 10^3$	Impact Strength*
Cast Resin	3-10	7-15	6-8	10-30	300-1,000		0.1-0.5
Moulding Material							
Wood-flour-filled	5-8	6-15	8-10	15-40	1,000-1,500	300-500	0.1-0.5
Chopped-cotton-fabric-filled	5-8	8-15	10-15	20-35	700-1,200	300-500	0.3-3.0
Mineral-filled	4-8	8-15	4-15	20-35	1,000-2,500		0.1-1.0
Laminated Material							
Paper-filled	8-25	15-30	5-12	20-40	1,000-3,000		0.2-2.0
Fabric-filled	8-20	15-30	5-12	30-45	500-1,500		1-5
Asbestos-filled	7-12	10-15	4-8	30-50	500-2,000		0.2-1.0

*Method of B.S. 771 for cast resin and moulding materials; B.S. 972 for laminated materials.

Reference: Ogorkiewicz, R.M. and P.D. Ritchie, Phenolic Resins, LONDON ILIFFE Books Ltd., 1967.

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APPENDIX G - EFFECTS OF NUCLEAR RADIATION DOSE RATE ON CABLE PERFORMANCE DURING A LOCA

More than 50 separate test reports on electrical cables were reviewed during the equipment environmental qualification evaluation. The major insulation materials used in the cable test samples were:

cross-linked polyethylene
chlorosulfonated polyethylene
ethylene propylene rubber
Neoprene
butyl rubber
silicone rubber.

(Proprietary flame-retardant additives and layered combinations of insulating materials and shields have also been used by various manufacturers to provide special features required by Licensees and their engineering contractors.)

Testing typically involved irradiation up to 200 Mrd at dose rates between 0.1 and 2.1 Mrd/h. Measurements of insulation resistance during the tests indicated that cable insulation resistance decreases with increasing dose rate, and that insulation resistance recovers after the exposure ceases. Typical reductions in insulation resistance are:

from 10^{11} to 10^{8} ohms at the low (0.1-0.25 Mrd/h) dose rates
from 10^{11} to 10^{5} ohms at the higher (1-2 Mrd/h) dose rates.

There are insufficient test data to determine the mathematical relationship between insulation resistance and dose rate. There is, however, test evidence that the dose rate effect combines with the pressure, temperature, humidity, and spray conditions to further reduce insulation resistance. For very high dose rates (i.e., greater than about 2 Mrd/h) during simulated LOCA conditions, insulation resistances in the range of 1000 to 10,000 ohms for 30 ft of cable (measured at 10 V dc) have been experienced.

During LOCA, the dose rates calculated in accordance with conservative NRC recommendations are typically 1 to 3 Mrd/h gamma and 10 Mrd/h beta during the first 10 hours of the LOCA. (These data are for a nominal 1000 MW(e) plant.) It can be seen that the dose rates for insulation subject to beta radiation exceed most test radiation dose rates by an order of magnitude.

There is concern, therefore, that exposed cables (i.e., cables not protected from beta radiation by cable tray covers or conduit) will not retain high enough insulation resistance to transmit reliable control and instrumentation signals without attenuation and distortion during the early stages (the first 10 hours) of a LOCA.

The Licensees of plants with exposed cables should carefully evaluate the possible effects of combined gamma and beta radiation dose rates, plus elevated temperature and moisture, on the ability of the cables to perform their functions. The evaluation should be based on available test data for the cables, or test data should be generated so that analysis can be performed.