TECHNICAL EVALUATION REPORT

FINAL DRAFT

EQUIPMENT ENVIRONMENTAL QUALIFICATION

CONSUMERS POWER COMPANY
BIG ROCK POINT PLANT

NRC DOCKET NO. 50-155

NRC TAC NO. 42515

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC TASK 198

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

January 31, 1981

NOTE — THIS IS A DRAFT

Because of schedule limitations, this report draft has not gone through the complete review cycle of FRC. While the overall conclusion is expected to remain the same, the reader is cautioned that some details of the report may change as the review is completed.

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

8102120202

CONTENTS

CONTENTS

CONTENTS

ACKNOWLEDGMENTS

The engineering, administrative, and editorial staff who produced this Technical Evaluation Report labored during long hours requiring intense attention to complex details. Without their extraordinary effort and devotion to this work, these reports could not have been produced in the short time that was available. The following are all to be thanked.

The principal contributors to the technical effort are:

J.C. Archer (ECI)
C.J. Crane
T.J. DelGaizo (W)
J.A. Murphy
G.J. Overbeck (W)
I.H. Sargent (W)
S.R. Schmitt
W.H. Steigelmann (SRC)

The associate technical contributors who supplied essential inputs are:

A. Cassell
C.B. Chan
M. Hargitay
J.E. Kaucher (W)
K. Kauffman
P.N. Noell
J.S. Scherrer (W)

The editorial staff who proofread and edited all the report drafts are:

R.J. Carelli
M. Dank
E.K. Friedman
P. Grant-Kingsberry
M.A. Musil
S. Reynolds
M. Rothman
M. Sherritze
R. Wilson

The word processing group who typed and made the numerous changes needed to arrive at the final report are:

F. Davis
A. Rogers
A. Oponski
A. McDonald

Overall management of the EEQ project was in the capable hands of Dr. S.P. Carfagno, FRC Project Manager, and Dr. Z. Zudans, the FRC Project Director.

The following oversaw and reviewed this work for the Nuclear Regulatory Commission:

J.J. Lombardo (Lead Engineer Equipment Environmental Qualification, Equipment Qualification Branch)

P. DiBenedetto (Section Leader, Equipment Qualification Branch)

Z.R. Rosztoczy (Chief, Equipment Qualification Branch)

E. Butcher (Project Officer)

There are many others too numerous to list who aided in this effort and they are all to be thanked.

Subcontractors

ECI = Energy Consultants, Inc.

SRC = Synergic Resources Corporation

W = WESTEC Services, Inc.

IDENTIFICATION OF PROPRIETARY INFORMATION

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

1. INTRODUCTION

1.1 PURPOSE OF THE REVIEW

The objectives of this Technical Evaluation Report are to evaluate nuclear power plant safety-related electrical equipment in accordance with criteria established by the NRC and to identify (1) equipment whose qualification documentation is adequate, i.e., substantiates that equipment is capable of performing its specified design basis safety function when it is exposed to a harsh environment and (2) equipment whose 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

Qualification criteria applied during the licensing of the older nuclear plants have been modified over the years, and 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, 382-72, and 381-77. 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 1E Safety Related Equipment." The positions documented therein are applicable to 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 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 within the scope of equipment addressed, definition of harsh environments, and adequacy of support 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 1E Electrical Equipment in Operating Reactors." (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. Originally it was intended that the Licensees would 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 equipment environmental qualification under SEP Topic III-12. FRC was given the assignment to review equipment environmental qualification documentation and to present the results in the form of a Technical Evaluation Report for each plant included in this program.

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 stated that the criteria to be used in evaluating the adequacy of the safety-related electrical equipment qualification are the DOR Guidelines. 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-0588 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 [85]* which included a second document, "Guidelines for Identification of That Safety Equipment of SEP Operating Reactors for Which Environmental Qualification Is To Be Addressed," [85] 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, [89] 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.

*For References, see Section 6.

1.3 SPECIFIC ISSUE BACKGROUND

On January 16, 1975, Consumers Power Company removed the Big Rock Point Plant from service after a design review of existing safeguards equipment revealed that they might be rendered inoperable to perform their function after the operation of a proposed new reactor depressurizing system. Many man-hours were expended during the subsequent six-month outage to qualify components in the two core spray systems, the two enclosure spray systems, and the containment isolation system (inside containment). The results of the qualification effort as well as the modifications made to the plant were provided to the NRC in Special Report No. 21. The NRC indicated their agreement with the actions taken by Consumers Power Company to qualify the equipment in a letter from K. R. Goller to R. B. Sewell dated May 30, 1975.

On December 23, 1977, in a letter from V. Stello to D. A. Bixel, the NRC requested that Consumers Power Company provide a submittal on the qualification of electrical equipment. The required response was provided on February 24, 1978, in a submittal from D. A. Bixel to D. K. Davis. An errata sheet to the February 24, 1978 letter was submitted on March 23, 1978 in a letter from W. S. Skibitsky to D. L. Ziemann.

On September 6, 1978, Consumers Power Company was advised by NRC in a letter from D. L. Ziemann to D. A. Bixel that information transmitted in previous submittals had been put into a standard format. Consumers Power Company was requested to complete the formatted tables (as more information was required than was contained in previous submittals) and to confirm accuracy and completeness of the information. On November 30, 1978, Consumers Power Company provided the NRC the requested information in a letter from D. A. Bixel to D. L. Ziemann.

In a letter from D. L. Ziemann to D. P. Hoffman dated February 15, 1980, NRC provided Consumers Power Company a set of guidelines to use in performing a review of the qualification of electrical equipment. Subsequently, on August 29, 1980, the NRC transmitted an amendment to the Big Rock Point operating license requiring that Consumers Power Company submit by November 1, 1980 information necessary to support a safety evaluation report on the

qualification of safety-related electrical equipment at the Big Rock Point Plant. This submittal contains the necessary information.

1.4 SCOPE OF THE REVIEW

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 whose environment is adversely affected by that event. Qualification aspects not included within the scope of this evaluation are:

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

- "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" [85]
- "Guidelines for Identification of That Safety Equipment of SEP Operating Reactors for Which Environmental Qualification Is To Be Addressed." [85]

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, [87] "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, with peak temperature and subsequent temperature/pressure profile as a function of time. If containment spray is to be used, the impact of the spray on required equipment should be accounted for.

The adequacy of a plant-specific profile is dependent 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 [87] 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 [87] 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 equivalent to the period from the initiation of the accident until the service conditions return to normal. Supplement 2 to IE Bulletin 79-01B [87] 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 [88] 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 (reference 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 that does not require pre-implementation review should be submitted to NRC for review by the implementation date.

- Qualification information for equipment required to achieve and maintain a cold shutdown condition should not be submitted later than February 1, 1981.

2.2.6 TEST SEQUENCE (DOR GUIDELINES SECTION 5.2.3)

Supplement 2 to IE Bulletin 79-01B [87] 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 programs 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 [87] 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 liquid systems inside and outside 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 (RCS = reactor coolant system):

	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)	NUREG-0588 (10/10/0 in RCS)
	or	
	NUREG-0578 (100/50/1 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."

3. METHODOLOGY USED BY FRC

The Licensee, Consumers Power Co., identified an extensive list of safety-related electrical equipment items in various locations of the Big Rock Point Plant in its submittals to the NRC. FRC analyzed the Licensee's listing and grouped together all identical equipment items located within plant areas that are exposed to the same environmental service conditions. This analysis resulted in a reduced listing containing (70) different equipment items that formed the basis for the review. 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 listing of safety-related electrical equipment items,* each equipment item was reviewed by FRC in relation to:

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

Topics not within the scope of FRC evaluation are:

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

---

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

The initial results of FRC's review of the equipment environmental documentation were issued to NRC as a Draft Interim Technical Evaluation Report (DITER) [86] on (October 21, 1980). Qualification data summary forms initially 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. [90] This information was analyzed by FRC to determine:

- what specific response was made to the FRC DITER
- whether any changes to the initial submittal were made by the Licensee
- what additional information was supplied (e.g., analysis, test report, or justification for qualification)
- whether any changes to environmental conditions were made
- whether any equipment had been 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. As requested by the NRC, suggested resolutions of deficient qualification areas and recommendations were provided where appropriate. Based on the FRC conclusion, each equipment item was assigned to one of the generic qualification categories provided by the NRC. NRC categories are described below.

The basic structure of each equipment item evaluation in Section 4 of the report follows 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 discussed in Section 4 of this report are associated with Reference 90.

NRC QUALIFICATION CATEGORIES AND DEFINITIONS

o NRC Category I.a
EQUIPMENT THAT FULLY 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 that the equipment has been found to be qualified for the life of the plant.

o NRC Category I.b
EQUIPMENT FOR WHICH DEVIATIONS FROM THE DOR GUIDELINES ARE JUDGED ACCEPTABLE

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.

o NRC Category II.a
EQUIPMENT THAT FULLY SATISFIES ALL APPLICABLE REQUIREMENTS OF THE DOR GUIDELINES AND HAS A SERVICE LIFE LESS THAN PLANT LIFE

This category includes equipment items that are fully acceptable on the basis that all applicable criteria defined in the DOR Guidelines are satisfied and that the equipment is qualified for a specific time interval which is less than plant life.

o NRC Category II.b
EQUIPMENT THAT FULLY SATISFIES ALL APPLICABLE REQUIREMENTS OF THE DOR GUIDELINES PROVIDED THAT SPECIFIC MODIFICATIONS ARE MADE

This category includes equipment items which will be fully acceptable and will satisfy all applicable criteria defined in the DOR Guidelines provided that specific modifications are made on or before the designated date. When the modifications are complete, the equipment can be considered qualified.

- NRC Category II.c
  EQUIPMENT FOR WHICH DEVIATIONS FROM THE DOR GUIDELINES ARE JUDGED ACCEPTABLE

This category includes equipment items which do not satisfy one or more of the applicable criteria defined in the DOR Guidelines; however, either sufficient bases have been presented to allow a determination that the specific deviations are judged to be acceptable for less than plant life or the specific deviations are judged by FRC to be acceptable for less than plant life based on review of the applicable qualification documentation associated with the general equipment environmental qualification program.

- NRC Category III
  EQUIPMENT THAT IS EXEMPT FROM QUALIFICATION

This category includes equipment items which are exempt from qualification on the basis that (i) the equipment does not provide a safety function (i.e., should not have been included in the equipment list submitted by the Licensee), (ii) the specific safety-related function of the component can be accomplished by some other designated component which is fully qualified, or (iii) the equipment is not subject to a potentially adverse environment as a result of accidents for which proper operation of the equipment is required. In addition, any failure of the exempt component must not degrade the ability of a qualified component to perform its required safety-related function.

- NRC Category IV.a
  EQUIPMENT WHICH 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. Specific justification for interim plant operation prior to testing should be provided for each item.

- 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. Specific justification for interim plant operation should be provided for each item.

- NRC Category V
  EQUIPMENT WHICH 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 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: (i) documentation reflecting qualification as specified in the DOR Guidelines has not been made available for review, (ii) qualification documentation made available for review is inadequate, or (iii) 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

4.1 METHODOLOGY USED BY THE LICENSEE

The final submittal of qualification documentation contained an introductory section which briefly described the Licensee's basic approach to qualification of equipment as required under the SEP Plant Topic III-12 and the DOR Guidelines.

4.1.1 COMPLETENESS OF THE EQUIPMENT LIST

The Licensee states the following regarding the development of the equipment list:

"The equipment list for the November 1, 1980 Big Rock Point Electrical Equipment Qualification effort was developed in three phases:

Phase I

1. Emergency Procedure EMP 3.3, Rev. 12, "Loss of Reactor Coolant," was reviewed to obtain a list of equipment used by operators during the accident.

2. The list was shortened by deleting equipment that wasn't located in a hostile area. The hostile areas are defined in Section A. [Appendix A of this report.]

3. Other equipment was deleted on a case-by-case basis as stated in Section B." [Appendix K of this Report.]

Phase II

1. The Instrument Data Book (Volume 4 of the Big Rock Point Plant Manual) was reviewed to obtain a listing of all equipment in the post-incident system, containment isolation system, the RDS system, the reactor protection system, the fire protection system, the emergency power system and the reactor vessel general system.

2. The list was then reduced based on whether or not the equipment was located in a hostile area, whether or not it was electrical and whether or not its function was deemed to be necessary to mitigate a LOCA or HELB.

Phase III

1. The lists obtained in the first two phases were then compared and the final list was generated. The list is found in Section C."

FRC concurs with the basic approach of the Licensee; however, no evaluation of equipment in non-harsh areas was accomplished, nor could FRC independently determine whether the areas are non-harsh. Appendix K contains FRC's comments about the equipment deleted from the list.

4.1.2 ENVIRONMENTAL SERVICE CONDITIONS

The Licensee states the following:

"Section E contains the environmental parameters for the five hostile areas." [Appendix A of this report.]

"Hostile Areas

Equipment considered for qualification was the equipment required to mitigate the effects of a LOCA or main steam line break (MSLB) inside containment or an MSLB outside containment and, in areas which suffer from the direct effects of these accidents, are defined as hostile areas. Areas of the Plant which house equipment required for the above accidents, but see only normal environment or a slight rise in room temperature due to operating equipment heat loss, are not hostile areas and are not considered here (as permitted per the NRC Regional Meetings on Electrical Equipment Qualification).

The five areas listed below are considered hostile for the purposes of electrical equipment qualification at Big Rock Point. Drawing 0740G40100 shows these areas.

1. The containment is subjected to pressure, temperature, humidity, gamma and beta radiation, submergence, containment spray and some thermal aging during the course of a main steam line break (MSLB) or a Loss of Coolant Accident (LOCA) inside.

2. The electrical penetration room is subjected to humidity during an MSLB outside of containment. During a LOCA inside containment, the electrical penetration room equipment receives a radiation dose due to shine from the containment wall. This room is unlike any of the other hostile areas outside of containment in that it is insulated and, therefore, suffers a temperature rise through the wall of containment.

3. The sphere ventilating room receives a gamma radiation dose due to shine from the containment wall during a LOCA. The construction of the building is such that heat can be readily dissipated; therefore, an ambient temperature is assumed. The structure is physically removed from the turbine building and will not be affected by an MSLB outside of containment.

4. The core spray room is a subterranean structure which houses the core spray pumps and core spray heat exchanger. The structure is physically removed from the containment wall and will, therefore, not be subjected to shine through the containment wall nor a temperature rise from heat transfer through the containment wall; however, when the containment sump water is recirculated through the heat exchanger subsequent to a LOCA, the recirculated fluid provides a gamma radiation source and a heat source. An MSLB outside containment has no effect on this room.

5. The pipe tunnel is located in the turbine building. It is an area where the fluid piping enters the containment building including the main steam and feedwater lines. Although some gamma radiation will be accrued by the equipment in the tunnel subsequent to a LOCA due to shine from the containment wall, the main steam line break provides a more severe environment to equipment in this area."

FRC considers that the Licensee approach is in accordance with DOR Guidelines. However, it is noted that LER-80-023 (transmitted to NRC on August 26, 1980) identifies possible temperature conditions in excess of 300°F for short periods of time which may affect the qualification of equipment.

Radiation:

The Licensee states:

"Radiation Inside and Outside Containment

The source terms used to calculate containment doses and all subsequent doses came from the ORIGEN code. The gamma and beta energies per disintegration for each nuclide are used. The beta energies are from the ORNL MEDLIST and the gamma energies were compiled from a listing for the computer program RASTUS. The RASTUS code calculates doses using Equation II as found in a letter from D. A. Bixel to J. G. Keppler dated May 11, 1977 regarding Consumers Power Company's response to IE Report 050-155/77-04."

"Containment

Containment beta and gamma dose rates for air and liquid were calculated at t = 0, 2 hours, 24 hours and 720 hours. Infinite cloud calculations were performed to obtain dose rates in all cases except the gamma dose rate in air.

The 30-day integrated radiation doses were obtained by segmenting time periods and summing the products of each time period multiplied by the average logrithmic dose rate in that time period. The results are as follows:

Containment Air Beta	1.32E+7 Rads
Containment Air Gamma	7.3E+5 Rads
Containment Water Beta	1.1.5E+6 Rads
Containment Water Gamma	2.83E+6 Rads

Beta doses were applied to cable only as the equipment provides adequate shielding for beta radiation."

"Electrical Penetration Room

The gamma dose rate due to containment shine in the electrical penetration room has been previously calculated. The methodology used was the same as referenced above (D. A. Bixel to J. G. Keppler, May 11, 1977). As was shown in that submittal, the electrical penetration room receives radiation from only 35% of the containment as a result of its location relative to the internal structures. Credit is also taken for shielding from the containment shell. The 30-day integrated dose was obtained by the same method described under containment doses. The shine dose at six feet from the containment surface is 5.21E+4 rads. The dose rates at six feet and on the containment surface were ratioed and multiplied by the 30-day integrated dose at six feet to obtain the 30-day integrated dose at the containment surface. The dose at the surface was found to be 7.6E+4 rads."

"Sphere Ventilating Room

The radiation dose due to gamma shine at the air shed was calculated by using the following three volumes of activity. One in the vicinity of the cooling unit, another in front of the clean-up demineralizer and a third at the top half of the containment sphere. Using the whole top half of the containment sphere is conservative since the shielding around the steam drum would diminish the dose rate. The computer program RASTUS was used to calculate the dose rates from three equivalent volume spheres. The two smaller spheres were taken to be of equal volumes and distances to the air shed. The radius of the smaller spheres is 3.81E+2 cm with a distance of 3.45E+2 cm to the air shed. The larger sphere has a radius of 1.47E+3 cm with a distance of 2.07E+3 cm to the air shed. Shielding used for all three shields is 3/4" of iron.

The dose rates form the three spheres as provided by RASTUS (runs 478 through 485) were added for each time period. The integrated dose is 1.94E+5 rads."

"Core Spray Room

The gamma radiation dose due to recirculating fluids in the core spray room was calculated. The calculation involved modeling the piping carrying radioactive sources in the room by using spheres of diameters slightly larger than the pipe diameter to account for all of the volume. The dose in the room is then a sum of the contributions from all of these sources. This calculation gives a 30-day integrated dose of 2.64E+3 rads; however, it assumed a 10% core melt and no noble gases or particulates in the liquid phase. Regulatory Guide 1.3 calls for 100% of the noble gases and 1% of the particulates in the liquid phase.

To update this dose to the Regulatory Guide 1.3 source terms, it was first multipled by ten for the 100% fuel melt case to give 2.6.4E+4 rads. To add the contribution from noble gases and particulates, a recent RASTUS run at t = 2 hours for containment air was used to compar the contribution from noble gases and particulates vs halogens. The total dose rate due to noble gases and in particulates was found to be roughly half of the total dose rate due to halogens (halogen dose rates were multiplied by 50 since the RASTUS run was for 1% of the halogens at t = 2 hours and the liquid phase contains 50% of the halogens)."

"The total 30-day integrated dose is therefore:

$$(2.64E+4) + (2.64E+4)\ (0.5) = 3.96E+4 \text{ Rads}$$

This is conservative since the dose rate of noble gas and particulates to halogens becomes less at times greater than two hours.

Doses from the containment atmosphere were not considered in this calculation due to the heavy shielding around the core spray room."

"Pipe Tunnel

Beta and gamma 30-day integrated doses were calculated for the turbine building given the following initial assumptions:

A. The entire primary cooling volume (3,830 ft^3 from Technical Specification 4.1.2) is dumped to the turbine building and it all flashes to steam.

B. The main steam isolation valve closes before any fuel failure occurs and no fuel failure occurs after closure."

FRC notes that the source terms of NUREG-0588 for radiation exposure calculations are required for both inside and outside containment dose. FRC

cannot independently determine that the noble gas, iodines, and fission products required by the Guidelines were used in the computer calculations.

4.1.3 AGING AND QUALIFIED LIFE

The Licensee presents an extensive discussion of aging, part of which is reproduced as follows:

> "Aging models and Technology - 'Aging' is a very complex yet universal phenomenon. The number of variables to be considered is very large, perhaps infinite, and a satisfying scientific approach that can be defended against all valid questions is clearly beyond the capability of today's state of knowledge. What is really desired in the present instance is a regression of equipment reliability during LOCA as a function of age. Each piece of equipment is constructed of various processes; has its unique demands on various characteristics of its materials of construction; fails to perform when varying numbers of material failures occur; and is exposed to its own unique history of environments and duty cycle in reaching a given chronological age at which various LOCA (or other accident or transient conditions) are postulated to occur. To produce the data for direct regression is clearly impossible so a very long layered process of assumption and approximation is employed. In most instances, these assumptions are imbedded in both the experiments from which the data are taken and the analysis of the data (regression). For instance, if temperature is chosen as the relevant environmental parameter, the equipment (material) is exposed to heat. The failure data (material characteristics) are regressed against temperature and time, and the time variable is equated to aging. In this process of severe simplification and layered approximation, conservatisms tend to creep in at every approximation in each layer (most of the experimental inaccuracy, lack of experimental control and data scatter go toward failure - the objective being to demonstrate what it can withstand vice what conditions will positively fail it). An excellent discussion of the variety of modeling techniques as well as a good example of the varying amount and quality of existing data are provided in EPRI Report NP-1558.
>
> In arriving at an aging model, the following approximations are usually made:
>
> a. Temperature is assumed to be the variable that controls aging.
>
> b. Materials defined by trade name or major chemical constituent are assumed the same at least when quoting references to existing test data and particularly when the results tend to be negative.
>
> c. Geometry of the part is given no special significance. When the data result from individual material tests, the samples are usually very thin (typically a few mils).

d. The way the material is exposed (direct or protected) is given no particular significance. In most tests, exposure is direct even though in practice the material is usually protected (air circulation, etc.).

e. Tests (especially isolated material tests) tend to equate a somewhat arbitrary reduction in a particular material characteristic (tensile strength, percent elongation, etc.) to failure. This characteristic may or may not be important to any specific application of the material. In any event, it is usually chosen very conservatively (sometimes as little as 25% reduction).

f. The Arrhenius equation is assumed to apply and to accurately relate damage at one temperature to damage at any other temperature. The basis for the Arrhenius equation appears to be as much or more empirical than it is theoretical. Judgments as to whether it should be applied in any given situation are very difficult to make. In most cases, its application leads to conservative results.

Due to the state of development of accelerated aging as a science, the models tend to be conservative. Very little comparison data between model predictions and actually aged power plant equipment exist. Therefore, it is difficult to improve models and remove conservatisms. The equipment to monitor actual in-service environment and equipment duty cycle would be extensive. Existing programs such as Nuclear Plant Reliability Data System (NPRDS) have no provisions to collect even estimates of such environmental data."

FRC has the following comments on the Licensee approach:

The Licensee has not adequately addressed the topic of qualified life for any of the safety-related equipment items. The DOR Guidelines require that the Licensee:

- establish (numerically) the qualified life for all equipment items containing components susceptible to degradation due to thermal and radiation environments
- 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
- replace equipment items (or components thereof) on a scheduled basis.

Qualified life is the maximum time period of normal service (during which the equipment may or may not be actively functioning) 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 DBEs. The qualified life of an equipment item may depend on the Licensee having implemented a specific preventive maintenance program to replace the components subject to aging-induced degradation. The qualified life value for a particular equipment item may be changed during its installed life where justified by new information.

Establishing the qualified life for equipment is a technically challenging task because of the paucity of reliable information concerning how materials are affected by the long-term exposure to the environment of a nuclear power generating plant. As is discussed more fully in EPRI Report 1558, with the possible exception of certain simple materials, FRC knows of no accurate, proven basis for extrapolating short-term (1 week to several weeks) oven-aging test results to an installed lifetime of 40 years. Technical knowledge regarding possible long-term synergistic effects of temperature, humidity, nuclear radiations, etc., does not exist.

FRC believes it is fundamentally unsound to specify that the qualified life shall be 40 years or any given value for every safety-related equipment item in the plant and then to attempt to prove that this "requirement" is met. The Guidelines do not require that the qualified life be shown to be 40 years (or any other value). Rather, they require that each Licensee provide evidence that the safety functions of the equipment in the plant can be performed adequately at all times.

FRC believes that a conservative qualified life should be established for each equipment item and that surveillance testing (necessary to monitor performance and identify degradation) should be performed to determine the need for maintenance or replacement. In this way, the qualified life of an equipment item can be extended as new information is gained.

4.2 EQUIPMENT QUALIFIED FOR PLANT LIFE

This section includes equipment items which are fully acceptable on the basis that (i) all criteria defined in Section 2 of the report are satisfied or (ii) sufficient data exist to determine acceptability by judgment.

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

This section 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. For the Big Rock Point plant, there are no items in this category.

4.2.2 NRC Category I.b
EQUIPMENT FOR WHICH DEVIATIONS FROM THE DOR GUIDELINES ARE JUDGED ACCEPTABLE

This section includes equipment items which do not satisfy one or more of the applicable criteria defined in the DOR Guidelines but for which sufficient information has been presented to determine that the specific deviations are acceptable. For the Big Rock Point plant, there are no items in this category.

4.3 EQUIPMENT QUALIFIED WITH RESTRICTIONS

This section includes equipment items fully acceptable on the basis that (i) all criteria defined in Section 2 of this report are satisfied, (ii) the equipment is qualified for a time interval less than plant life, or (iii) the equipment requires specific modification which, when completed, will establish full qualification.

4.3.1 NRC Category II.a
EQUIPMENT THAT FULLY SATISFIES ALL APPLICABLE REQUIREMENTS OF THE DOR GUIDELINES AND HAS A SERVICE LIFE LESS THAN PLANT LIFE

The following equipment items are fully acceptable on the basis that all applicable criteria defined in the DOR Guidelines are satisfied and that the equipment is qualified for a specific time interval which is less than plant life.

4.3.1.1 Equipment Item No. 1
Electrical Penetrations Located Inside Containment
Model Not Stated
(Licensee References 1, 2, 36, 78, 80, 4-5, 4-65, 4-66)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Guidelines require that in order to determine the adequacy of the qualification of equipment, it is necessary to specify the service environment to which the equipment is exposed during normal and accident conditions. Qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. Furthermore, the Guidelines require the model of the test specimen be the same as that of the equipment being qualified. The type test is valid only if the installed equipment and test specimen have the same design, materials, and production procedures. An analysis of the impact of deviations between the test specimen and the installed equipment is an essential part of the qualification documentation.

Reference 4-5 is a Licensee memo which provides an evaluation of the similarity between penetrations which were subjected to tests and penetrations

installed in the plant.

Reference 4-66 is a letter to NRC containing design and test information on electrical penetrations and providing a safety evaluation of the installed penetrations. FRC has reviewed these references and notes the following:

a. The pressure/temperature steam tests envelop the LOCA conditions and are judged to be acceptable.

b.

c. The testing and evaluations do not provide information on irradiation, aging, chemical spray, humidity, or the effects of heating due to short circuit currents which would be applicable due to the number of non-safety-related electrical items in containment.

It should be noted that FRC has reviewed test data where some formulations of epoxies and resins were damaged by radiation (at low levels), thermal aging, and humidity.

It is suggested that the Licensee determine from the designer of the penetrations whether any testing complying with the Guidelines requirements or IEEE standards has been conducted on penetrations having the same materials as are installed at Big Rock Point.

LICENSEE RESPONSE:

Component Electrical Penetrations (Types 1 Thru 7)

Type 1: 6" penetration assembly embedded with 20 feet long, 92 single conductor #14 AWG solid copper wires with GE versotol insulation and geoprene jacket.

Type 2: 6" penetration assembly embedded with 20 feet long, 52 single conductor #14 AWG solid copper wires with GE versotol insulation and geoprene jacket.

Type 3: 6" penetration assembly embedded with 20 feet long, 52 single conductor #14 AWG solid chromel & alumel wires with neoprene insulation firmly bonded to the conductor.

Type 4: 6" penetration assembly embedded with 20 feet long, 20 single conductor #8 AWG solid copper wires with GE versotol insulation and geoprene jacket.

Type 5: 6" penetration assembly enbedded with 20 feet long, 6 single conductor #4/0 AWG 19 strand copper with Okonite oil-based rubber insulation and heavy-duty black neoprene jacket.

Type 6: 6" penetration assembly embedded with 5 feet long, 4 copper rods with 1/2" outer diameter and other insulating materials.

Type 7: 1-1/2" penetration assembly embedded with 20 feet long, 8 single conductor #14 AWG solid copper wires with GE versotol insulation and geoprene jacket.

The subject electrical penetrations are used for Class 1E as well as non-Class 1E circuits inside the containment. This qualification report, therefore, includes both the functional and the pressure boundary integrity of these penetrations.

1. Electrical Penetration Assemblies

Penetration Types 1 thru 6 consist of 6" nipples, 12" long with 6" conduit steel bushing at one end and a 6" to 8" concentric reducer welded at the other end. After fabrication of the complete penetration as per the procedure given below, the complete unit as such is welded to 8" Schedule 40 sleeve 12" long which in turn is welded to reactor building shell.

Penetration Type 7 consists of 1-1/2" nipple, 9" long with 1-1/4" nipple at one end and a 1-1/2" to 3" concentric reducer welded at the other end. After fabrication of the complete penetration as per the procedure given below, the complete unit as such is welded to 3" Schedule 40 sleeve which in turn is welded to lock shell.

a. Fabrication Procedures: Fabrication of penetration units from 6" or 1-1/2" Schedule 40 nipples is performed in the following sequence:

(i) Wire brush inside wall of the 12" long 6" nipple (or 1-1/2" nipple for Type 7) and swab with toluene solvent.

(ii) Thread the wire, cable or copper rod into the spacers and insert it into the nipple until Spacer Disc 1 reaches the bushing (or the end of the 1-1/4" nipple). Place the unit in the vertical position such that Spacer Disc 1 is resting on the bushing (or the end of 1-1/4" nipple) and is level.

(iii) Place a suitable length of thin wall conduit or other suitable tube through the center hole of Spacers 2 through 5 and insert tube to near bottom spacer. Pour a 2-3/4" to 3-1/8" layer of Chico "A" compound taking care not to splash any of the compound on the inside wall of the pipe above this layer. After completing the pour, withdraw the tube and move Spacer 2 into into the nipple until small amount of Chico "A" begins to appear through various holes and outside edge of spacer. The Chico "A" is allowed to set for 3 days.

(iv) Remove all foreign matter and swab internal pipe surface with solvent. Pour 1/2" to 5/8" layer of epoxy resin (with glass bead filler) on top of the Chico "A" compound in the same manner that Chico "A" was poured. After completing the pour, withdraw the tube and move Spacer 3 into the nipple until small amounts of resin begin to appear through the various holes and outside edge of the spacer. The resin shall be allowed to set for a least 24 hours.

(v) Pour 3" to 3-1/2" layer of Ozite "B" compound in the same manner as previous pours after first bringing the temperature of the Ozite to approximately 350°F. Spacer 4 is then moved into the nipple until small amounts of Ozite begin to appear through various holes and outside edge of the spacer. The Ozite "B" compound is allowed to set for at least 12 hours.

(vi) Pour 2-1/1" to 3-1/8" layer of Chico "A" compound in the same manner as previous pours. On 1-1/2" units only move Spacer 5 into the nipple until small amounts of Chico begin to appear through the various holes and outside edge of the spacer. Allow the Chico "A" to set for 3 days.

(vii) For 6" units only, pour a 1/2" to 5/8" layer of epoxy resin on top of Chico in the same manner as previous pours. Spacer 5 shall then be moved into the nipple unit small amounts of resin begin to appear through the various holes and outside edge of the spacer. After the spacer is in position, pour approximately 1/4" float of resin over the top of spacer so that it is completely covered. The resin is allowed to set for at least 24 hours.

(viii) For 6" units, a 16" extension nipple is also attached to the penetration unit to provide a raceway for carrying the conductors to the tray system and to permit, in combination with the fiber barrier, the addition of a thermoplastic compound to increase further the reliability of the penetration.

2. The qualification information of the subject electrical penetrations is based on the following documents and discussions:

a. Document Reference 1 includes Pittsburgh Testing Lab (PTL) Report 1-C dated March 13, 1958 and Report 3-B dated April 17, 1958. These test reports cover actual leakage rate test and LOCA tests on similar design penetrations made by General Electric. The similarity between electrical penetration Types 1 thru 7 and the penetration assembly specimens covered by PTL reports is based on the following facts:

(i) Production procedures used for the assembly of both types of penetrations are same.

(ii) Both types of penetrations are designed by General Electric.

(iii) Similar insulating materials such as Textolite, Chico, epoxy and Ozite "B" have been used for the assembly of both types of penetrations.

(iv) Penetration Type 1 which consists of 92 single conductor #14 AWG solid copper is similar to 3 units of 92 single conductor #12 AWG solid copper tested by PTL.

(v) Penetration Type 5 which consists of 6 single conductor #4/0 AWG stranded copper used in 3 kV circuits is similar to 2 units of 6 single conductor #4/0 AWG for 5 kV circuits tested by PTL.

(vi) Penetration Types 2, 3, 4 and 7 are similar to 92 single conductor #12 AWG solid copper tested by PTL.

During environmental qualification testing covered by PTL Reports 1-C and 3-B, the penetration assemblies were first subjected to air pressure of 60 psig (74.7 psia) for one hour and then subjected to steam exposure tests at 307°F, 60 psig (74.7 psia) and 100% relative humidity for 23-1/2 hours. At the end of the test, the units were cooled to room temperature and successfully passed a 150 psig cold hydrostatic pressure test for one minute without any leakage.

b. Document Reference 2 includes Specification E-2345-104 for test of typical sphere electrical penetration assembly units prepared by Bechtel Corp. dated February 6, 1958. This specification covers the qualification testing of electrical penetration units by Pittsburgh Testing Laboratory described in Subparagraph 2.2.1 above.

c. Analysis of materials used in the fabrication of complete penetration, as covered by Document References 80 and 36, confirms the following:

(i) That the properties of Chico "A" compound are similar to Portland Cement which, being inorganic in nature, is not subject to radiation aging.

(ii) That Textolite spacer discs used in the penetration assembly are similar to epoxy or phenol formaldehyde laminates with glass fabric and possess radiation and thermal withstand properties which are equal to or better than regular epoxy resins. Per Document Reference 36, this type of material can withstand up to 8.3×10^7 R as given in Table 3.22 on Page 156.

(iii) That the epoxy compound used in the penetration assembly is good for maximum continuous service temperatures up to 400°F.

d. Document Reference 36 on Page 96 indicates that epoxy resins can withstand radiation dosage up to 9.5×10^8 rads without degradation of physical or electrical properties.

3. Discussions:

a. Radiation:

The radiation withstand capability of the complete electrical penetration is based on the following materials:

(i) Epoxy - As discussed in Paragraph 2.2.4, it is confirmed that epoxy can withstand radiation dosage up to 2.5×10^8 R which meets the requirement of 1.373×10^7 R.

(ii) Textolite - Paragraph 2.2.3(b) confirms that Textolite is similar to epoxy laminate with glass fabric and possesses radiation withstand properties even better than 2.5×10^8 R and meets the requirements of 1.373×10^7 R.

(iii) Chico "A" - Paragraph 2.2.3(a) confirms that Chico "A" is similar to Portland Cement which being inorganic in nature is not subject to radiation damage and, therefore, meets the requirements of 1.373×10^7 R.

(iv) Ozite "B" - Ozite "B" is an asphaltic based hydrocarbon manufactured by G&W Electric Specialties Corporation. Reference 5 document, Pages 456 through 459, indicates that such hydrocarbons are not affected at radiation levels below 4.5×10^8 rads.

b. Simulated service conditions and test duration:

Based on the following facts, it is concluded that the LOCA test duration of 23-1/2 hours (Document References 1 and 2) covers the entire time until the conditions return to essentially ambient.

(i) Analysis of containment parameters indicates dwell at maximum temperature of 235°F for 1 hour whereas during the actual LOCA test maximum temperature of 307°F was maintained for 1 hour.

(ii) Analysis of containment parameters indicates dwell at maximum pressure of 41.7 psia for 1 hour whereas during the actual LOCA test pressure of 41.7 psia was maintained for all the 23-1/2 hours.

Further, it is added here that even though the test duration requirement is 30 days, the analysis of containment parameters shows that following a LOCA, temperature and pressure conditions return to ambient in about 3 days.

c. Aging:

Based on the very high thermal and radiation withstand capabilities (listed below) of various components used in the fabrication of complete penetration assembly, it is concluded that these penetrations assemblies are not susceptible to thermal aging and are considered to be qualified for 40-year life requirement at Big Rock Point Plant.

(i) Epoxy and Textolite (Laminated epoxy) - Suitable for continuous operation at service temperatures of 400°F and can withstand radiation dosage of 2.5×10^8 rads.

(ii) Chico "A" - Being similar to Portland Cement is not susceptible to radiation or thermal aging.

FRC EVALUATION:

The data provided by the Licensee submittal resolves the questions raised in the DITER by analysis of the radiation effects on the materials. However, FRC is not aware of any proven method of simulating and accelerating all of the factors involved in aging which would establish a 40-year qualified life.

FRC CONCLUSION:

This item belongs in NRC Category II.a. The Licensee should perform detailed surveillance testing which would assure both electrical and mechanical integrity and identify any degradation which would indicate the need for maintenance or replacement.

4.3.1.2 Equipment Item No. 13
Motorized Valve Actuators Located in the Core Spray Room

Actuates Post-Incident HX Cooling Water Valve (MO-7066)
(Original Licensee References 4-12, 4-15, and 4-46;
Final Licensee Reference 32)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

Reference 4-12 is a test report for two motorized valve actuators ( ), and two by One actuator from each manufacturer was dc powered and the other ac powered. All four test units had been removed from the Big Rock Point Nuclear Power Plant and decontaminated prior to the test. Reference 4 states that the valves are located outside of containment, are not subject to harsh environment spray or submergence and that the radiation dose resulting from LOCA is 0.13 Mrad.

FRC has reviewed Reference 4-12 and notes the following:

a. The Guidelines require that the test specimen be the same as the equipment being qualified. The installed and tested units are the same since the test units were originally installed at the Big Rock plant.

b. The reported test was quite comprehensive (including load cycling and a 36-hour steam/28-hour water spray exposure), and the test unit's performance was satisfactory. The temperature and pressure profiles used in the test program adequately envelop the plant-specific conditions stated in Reference 4.

c. The actuator was not subjected to a documented pre-aging prior to the steam/spray exposure. The Guidelines state that thermal aging of a test specimen is required unless the unit contains no materials susceptible to significant degradation due to aging. The qualified life for these units should be established.

d. The actuator was not exposed to nuclear radiation to simulate LOCA conditions. However, information was submitted to demonstrate that the materials used would not be degraded by exposure to nuclear radiations (as is required by the Guidelines) in Reference 4-46.

Based on the above, FRC considers the valve operators to be satisfactory, subject to confirmation that changes in materials discussed in Subsection 3.3.2.2 have been accomplished.

LICENSEE RESPONSE:

This actuator contains a ac motor with Class B insulation. It is located outside containment in the post-incident room. The valve is actuated manually in the control room when recirculation of the containment sump water begins. The valve opens the cooling water flow path for the post-incident heat exchanger cooling from the fire water system.

The valve, being in the post-incident room, does not experience a harsh environment until after recirculation begins and the room heats up and radiation shine from the sump water is present. Because the valve is operated while the area remains at normal conditions, it is qualified for operation. Once opened, it will not be necessary to close the valve. In the event it cannot be opened remotely, this valve, or a bypass, can be operated manually in the post-incident room.

FRC EVALUATION:

Reference 32 is the same as Reference 4-12 and was discussed in the DITER and in the Licensee Response. Because it did not include accelerated aging or nuclear radiation exposures, this test program provides useful, but not conclusive, evidence that these MVAs will function adequately. The Guidelines require that qualification be demonstrated for a minumum of 1 hour under accident conditions. Although the Licensee has not addressed the deficiencies in the test program for this equipment item FRC believes that sufficient analysis has been provided in the Licensee Response for Item Nos. 11A and 11C to eliminate the concern with regard to radiation exposure. However, it is preferable that the Licensee provide an explicit analysis. Also, the Licensee should make a conservative estimate of the qualified life, and develop a maintenance and equipment performance surveillance schedule to ensure that this equipment will perform properly under accident conditions. It is advisable to renovate the installed units, replacing the nonmetallic components.

FRC CONCLUSION:

This equipment belongs in NRC Category II.a because the Licensee has provided sufficient evidence to show that the basic requirements of the Guidelines are satisfied. Because the equipment may have reached, or may be

soon reaching, the end of its qualified life and the qualification documentation is not as comprehensive and thorough as is generally desired, the Licensee should consider replacing or rebuilding the installed units.

4.3.1.3 Equipment Item No. 15
Valve Actuator Located in the Core Spray Room

Actuates Backup CS HX Valve (MO-7072)
(Original Licensee References 4-12, 4-15, and 4-46;
Final Licensee Reference 32)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

Reference 4-12 is a test report for two motorized valve actuators (having Class B insulation), and two by . One actuator from each manufacturer was dc powered and the other ac powered. All four test units had been removed from the Big Rock Point Nuclear Power Plant and decontaminated prior to the test. Reference 4 states that the valves are located outside of containment, are not subject to harsh environment spray or submergence, and that the radiation dose resulting from LOCA is 0.13 Mrad.

FRC has reviewed Reference 4-12 and notes the following:

a. The Guidelines require that the test specimen be the same as the equipment being qualified. The installed and tested units are the same since the test units were originally installed at the Big Rock plant.

b. The reported test was quite comprehensive (including load cycling and a 36-hour steam/28-hour water spray exposure), and the test unit's performance was satisfactory. The temperature and pressure profiles used in the test program adequately envelop the plant-specific conditions stated in Reference 4.

c. The actuator was not subjected to a documented pre-aging prior to the steam/spray exposure. The Guidelines state that thermal aging of a test specimen is required unless the unit contains no materials susceptible to significant degradation due to aging. The qualified life for these units should be established.

d. The actuator was not exposed to nuclear radiation to simulate LOCA conditions. However, information was submitted to demonstrate that the materials used would not be degraded by exposure to nuclear radiation (as is required by the Guidelines) in Reference 4-46.

Based on the above, FRC considers the valve operators to be satisfactory subject to confirmation that changes in materials discussed in Subsection 3.3.2.2 have been accomplished.

LICENSEE RESPONSE:

> This actuator contains a dc motor with Class B insulation and weatherproof enclosure. It is located outside containment in the post-incident room. The valve is actuated manually in the control room only in the event of a failure in the fire header system supplying core and containment spray water. It is only an emergency source of water and will not be normally used. In the event it is opened, it will be closed within 21 hours when recirculation begins. It will not have to operate in a harsh environment and will be maintained closed throughout the incident.
>
> Because the valve operates prior to the area experiencing higher than normal temperatures and radiation doses, the actuators are qualified for operation. Even in the event the valve actuator will not operate remotely, the valve can, if necessary, be operated manually in the post-incident room prior to recirculation.

FRC EVALUATION:

Reference 32 is the same as Reference 4-12 and was discussed in the DITER and in the Licensee Response. This test program provides evidence that these MVAs will function adequately. The Licensee should make a conservative estimate of the qualified life and develop a maintenance and equipment performance surveillance schedule to ensure that this equipment will perform properly under accident conditions.

FRC CONCLUSION:

This equipment belongs in NRC Category II.a because the Licensee has provided sufficient evidence to show that the basic requirements of the Guidelines are satisfied. Because the equipment may have reached, or may soon be reaching, the end of its qualified life, the Licensee should consider replacing or rebuilding the installed units.

4.3.1.4 Equipment Item No. 22
Pressure Transmitter Located Inside Containment

Reactor Vessel Pressure (PT-IA07C)
(Original Licensee Reference 4-29; Final Licensee References 21 and 47)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

Reference 4-29 is a certificate of compliance document from Rosemont which is not acceptable documentation according to the DOR Guidelines. The Licensee did not provide environmental qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

Specific information relative to the transmitter's location and elevation inside containment was not provided, and it is possible (according to a reference document) that the instrument will be subjected to submergence. The Licensee should review this item and provide qualification for the submergence condition if flooding is possible.

FRC has reviewed its files to determine if previous testing had been performed on the pressure transmitter. This search yielded information pertaining to pressure transmitter testing on a potentially similar model transmitter; however the degree of similarity will have to be evaluated by the Licensee and resubmitted to FRC since the test was conducted

for the Bailey Meter Co. of Wickliffe, Ohio and is proprietary. The Licensee must establish that the same materials and components exist in the pressure transmitter at Big Rock Plant and the transmitter which was type tested in order to establish similarity.

FRC concludes that the Licensee should submit appropriate qualification documentation and address the submergence issue.

LICENSEE RESPONSE:

Reference 47 stated that the installed transmitter meets the nuclear qualification data with respect to radiation, vibration, temperature and pressure as verified in (Reference 21). The pressure transmitter tested in this report was 1152DP4A22. The transmitter first underwent thermal aging. The thermal aging test was two temperature cycles of All temperatures were sustained for a minimum of one hour. After each hour of stabilization, readings were taken. This was done in accordance with IEEE-323 (1971). Next, the transmitter was exposed to 1×10^6 rad/h for five hours. Data was taken at each hour. Next, the unit was seismically tested and then environmentally tested. The test sequence was (1) to admit hot air to raise the conditions in the test chamber to 350°F and 60 psig for 10 minutes, (2) admission of saturated steam raising the temperature and pressure to 316°F and 70 psig holding for 1 hour, (3) to reduce to 303°F and 55.4 psig and holding for 7 hours and (4) to reduce to 230°F and 6 psig and holding for 42 hours. These environmental conditions exceed that required to demonstrate qualification. The test duration is a total of 50 hours which does not cover the time at which ambient conditions are reached at Big Rock Point (72 hours); however, the duration is considered sufficient because the test conditions greatly exceed the environmental conditions following a LOCA at Big Rock Point. The radiation levels are in excess of those required to demonstrate qualification. The transmitters are encased in a housing so that the additional dose due to beta radiation is negligible. The test did not include spray. The spray at Big Rock Point consists of Lake Michigan water. Since this transmitter is enclosed and passed the radiation and environmental tests, sprays will not product any adverse effects. Further, this transmitter is located in a room such that it will not be subjected to containment sprays directly. No further aging information could be located other than the aging test presented. This test, coupled with the fact that the transmitters are in a normal environment of 50°F to 90°F (seasonal variation) and the arguments presented in Section IV, provides qualification of aging.

The transmitters underwent an aging test, then a radiation test and then an environmental test and met the acceptance criteria placed on the operation both during and after each test. The transmitters are enclosed

and, therefore, not affected by the spray. Also, they are shown to be qualified for aging. Therefore, these transmitters are considered to be fully qualified.

FRC EVALUATION:

In response to the DITER, the Licensee has provided References 21 and 47 as evidence of qualification. FRC has reviewed these documents and notes the following:

a. Reference 47 is a Quality Certification of Compliance Data Sheet which certifies that the model number is and that Reference 21 is the applicable test report.

b. Reference 21 describes testing conducted on a Model serial number 090 pressure transmitter. The test sequence was: thermal aging, radiation testing, seismic testing, and LOCA testing.

FRC also notes:

- The thermal aging consisted of subjecting a complete transmitter to a sequence of 1-hour (minumum) exposures to different temperatures (two cycles of 100-0-200-100°F; total time about 16 hours). The unit's calibration was checked at six points between 0 and 100% of full scale at each temperature. The results showed negligible drift in the output signal. While this test does demonstrate that the stability of this equipment is good, it is not translatable into an equivalent period of installed life in the plant. The qualified life has not 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 required by the Guidelines.

- The transmitter was exposed to an integrated dose which exceeded the postulated accident dose. In addition, the accuracy was well within specification. FRC finds this acceptable.

- The steam temperature/pressure profile in the referenced test did not fully envelop the plant/specific profile, but the deviation (pressure) is judged to be acceptable. The test duration (50 hours) was more than adequate to envelop the required 41-hour accident duration at Big Rock Point (see Appendix A). In addition, the peak temperature and pressure of the test chamber far exceeded the required profile for an 8 hour time interval. However, after the 8th hour, the test chamber pressure (6 psig) was lower than the required profile pressure (approximately 24 psig). This is not considered significant because the transmitter was exposed to parameters far in excess of the required values and pressure integrity remained intact. FRC finds the LOCA testing satisfactory. In addition, the accuracy is acceptable.

- With respect to relationship to the test specimen, the report states:

  "All qualification tests for radiation, seismic vibration, and steam-pressure w e passed by the test transmitter. The (differential pressure) is representative of the other models of pressure transmitters AP (absolute), GP (gage) and HP (highline) in mechanical and electrical features. The only differences in the various models is the input (pressure) and output (electrical) relationships. These are primarily calibration differences; not influencing qualification capability. FRC finds the relationships to the test specimen satisfactory."

- The Licensee states that the transmitter is located in a room where it is not directly subject to containment spray. FRC concludes that qualification for spray is not required.

FRC CONCLUSION:

This equipment item belongs in NRC Category II.a because all applicable criteria defined in the DOR Guidelines are satisfied with the exception of the limiting condition that aging and qualified life have not been adequately addressed.

4.3.1.5 Equipment Item No. 41
Terminal Connections Located Inside Containment

(Original Licensee Reference 4-6; Final Licensee Reference 27)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

FRC has reviewed the information provided by the Licensee and has the following conclusions and comments:

a. The units tested and discussed in Reference 4-6 are designated as

The Guidelines state that the test specimen must be the same as the equipment being qualified. The Licensee did not present evidence that the test specimen is identical to the installed equipment. In addition, the Licensee did not present an analysis comparing the impact of deviations between the test specimen's and 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 two units are similar.

b. The tests conducted and reported in Reference 4-6 would envelop the LOCA or MSLB conditions identified by the Licensee for the Big Rock Point Plant.

c. The Licensee submittals do not identify whether the terminal connections are exposed or protected in the plant and in the test environment.

The Licensee should provide evidence that the terminals installed in the plant are the same type as tested and reported by Reference 4-6. If the installed and tested items are not the same, an analysis as specified in the Guidelines should be provided.

LICENSEE RESPONSE:

Component Terminal Connections - RDS

These terminal connections are used in the RDS system and were tested by AMP and Franklin Institute Research Laboratories (Reference 27).

The test specimen is Size 22-16 terminals having polyvinylidene fluoride insulating sleeves. This is the same as was installed at Big Rock Point.

The specimens were exposed to radiation at the rate of 0.5 to 1.0 Mrads/h for an accumulated dose of 2×10^8 rads. The specimens were placed in a test chamber preheated to and then steam was added to the chamber. The temperature The pressures corresponded to saturation. A chemical spray solution of boric acid (3,000 ppm) and NaOH (pH of 10.5) was injected 19 seconds after steam started. The sprays used are more caustic than required to demonstrate qualification. The conditions were increased to 352°F and 121 psig, held for 3 hours and then increased to These conditions were held for 2 hours, reduced to 253°F and 14 psig and held for 4 days. The chamber was then opened. The terminal connections passed the axial load and dielectric withstand voltage tests conducted after the irradiation and environmental tests. The test conditions enveloped the required environmental conditions both at the peaks and duration. No thermal aging qualification data has been obtained. The connections were installed in 1975 and are in a normal environment of 50-90°F (seasonal variation).

The effects of aging on the materials of construction will be considered by June 30, 1982. Due to the mild ambient temperature and the newness of the materials, these terminal connections are considered acceptable for use in the interim period.

The terminal connections are considered qualified as they passed an environmental test whose conditions were far more severe than those required to demonstrate qualification. The test duration encompassed the time to ambient (3 days for Big Rock Point). The test specimens were similar. The terminal connections are evaluated to have sufficient life.

FRC EVALUATION:

a. The identity problem described in the DITER has been resolved on page 174 of the Licensee submittal which states that Size 22-16 terminals are installed in the plant.

b. It is noted that pre-aging was not included in the test and that the Licensee will evaluate aging prior to June 1982. In this regard, FRC is not aware that there is any proven method of simulating and accelerating all of the factors which would affect time-dependent degradation (aging).

FRC CONCLUSION:

This item belongs in NRC Category II.a. The Licensee should establish a conservative qualified life and perform the surveillance testing necessary to monitor performance and identify any degradation which would indicate the need for maintenance or replacement.

4.3.1.6 Equipment Item No. 46
Electrical Cables Located in the Electrical Penetration Room

(Original Licensee References 4-9, 4-10, and 4-11; Final Licensee Reference 35)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Guidelines require that the model of the test specimen be the same as that of the equipment being qualified. The type test is valid only if the installed equipment and test specimen have the same design, materials and production procedures. An analysis of the impact of deviations between the test specimen and the installed equipment is an essential part of the qualification documentation. In a case where cable model numbers are not usually applicable, however, the manufacturers' formulations and identification of the insulation and jacket materials are considered to be an acceptable alternative to a model or part number. For example, the

descriptions provided on sheet number 516726 Paragraph B of Reference 4-9, page 2-1 of Reference 4-10, and page 2 of Reference 4-11 provide adequate identification of the cable.

FRC has reviewed the referenced documentation and notes the following:

a. The Licensee submittals do not identify the installed cables in a manner which permits independent verification that the type cable installed is the same as the type tested. As noted during the site visit, the Licensee should identify the installed cables in such a manner that it can be established

b. Regarding Note 1 of Reference 4, the Guidelines state that specifying saturated steam as a service condition during type testing of equipment that will become flooded in service is not an acceptable alternative for actually flooding the equipment during the test. Items subjected to steam and spray need to be tested under steam and spray conditions. Items subjected to submergence need to be tested under conditions of submergence.

It is suggested that the Licensee contact the manufacturers to determine whether there are tests of submergence applicable to the installed cables.

LICENSEE RESPONSE:

1. 600 Volt Class Cables, Sizes as Detailed Below, With XLPE (Radiation Cross-Linked Polyolefin) Insulation and Jacketing Material

   - 2/C #14 AWG
   - 3/C #14 AWG
   - 3/C #16 AWG
   - 6/C #14 AWG
   - 12/C #16 AWG
   - 2/C #16 AWG Twisted and Shielded
   - 2/C #16 AWG Twisted

2. 1000 Volt Class Cables, Sizes as Detailed Below, With XLPE (Radiation Cross-Linked Polyolefin) Insulation and Jacketing Material

   - 2/C #12 AWG
   - 2/C #10 AWG
   - 2/C #12 AWG
   - 3/C #10 AWG

The subject cables are used in Class 1E circuits outside the containment in areas not subjected to submergence or water spray. The qualification is based on the following documents and discussion:

A. Document Reference 35 demonstrates that the cable with identical insulation is qualified for use inside the containment based on 40-year life and LOCA. In containment, environmental conditions at Big Rock Point are far more severe than the conditions outside the containment. The test cycle included the following:

1. Thermal preaging at 150°C (302°F) for and at 160°C (320°F) followed by thermal and radiation aging at 150°C (302°F) and 5×10^7 R for

2. Combined LOCA and radiation exposure as below:

   - Radiation 1.5×10^8 R.

   - Temperature and pressure as follows:

   177°C (351°F) at 84.7 psia for 10 hours; 135°C (275°F) at 45.7 psia for 4.5 days; 100°C (212°F) at 24.7 psia for 26 days.

   The chemical spray during LOCA consisted of 3,000 ppm of boron and 0.064 molar sodium thiosulfate and adjusted with sodium hydroxide to a pH value of between 9.5 to 11.

3. Dielectric test for five minutes while immersed in water.

It is concluded that the temperature, radiation, humidity and aging requirements have been exceeded by a substantial margin and, therefore, the cable is qualified for this appplication.

FRC EVALUATION:

FRC concludes the report establishes the environmental qualification of this equipment item according to the requirements of the Guidelines except for submergence. This conclusion does not include 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. However, the thermal aging exposure and the simulated LOCA exposure are both very severe. 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.

FRC CONCLUSION:

This equipment belongs in NRC Category II.a. Since, FRC is not aware of any proven method of simulating and accelerating all of the factors which affect time-dependent degradation (aging) of non-metalic materials, the Licensee should establish a conservative qualified life and perform the surveillance tests necessary to monitor performance and identify any degradation which would require maintenance or replacement.

4.3.1.8 Equipment Item No. 47
Electrical Cables Located in the Sphere Ventilation Room
Rockbestos Model Firewall III
(Original Licensee Reference 52; Final Licensee References 91, 44, 46

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE STATEMENT:

Component Cable Type 5

Type 5 - 2/C #14 AWG, XLPE/NEO, Firewall III, 600 V

A. The above cable is a replacement to the existing cables outside the containment in areas not subjected to submergence or water spray.

B. Document Reference 52 demonstrates that the cable with identical insulation is qualified (without taking any credit for the jacket) for use inside the containment based on 40-year life and LOCA. In containment, environmental conditions are far more severe than the conditions outside the containment. The test cycle included the following:

1. Thermal aging at 150°C (320°F) for 1,300 hours (against the requirements of 850 hours to simulate a 40-year life); gamma radiation from a Cobalt-60 source for a total dosage of 2×10^8 rads; a LOCA simulation test with chemical spray at a peak temperature or 346°F; 100% relative humidity and a peak pressure of 113 psig. Total duration of test was 30 days.

2. After completion of a LOCA test, further thermal aging at 200°F and 100% relative humidity for 100 days.

3. Dielectric test for 5 minutes at 80 V ac/mil was performed after LOCA and thermal aging tests while immersed in water.

C. No credit has been taken for the neoprene jacket furnished. It is concluded that the temperature, radiation, humidity and aging requirements have been exceeded by a substantial margin and, therefore, the cable is qualified for use in this application.

FRC EVALUATION:

Licensee Reference 91 is a manufacturer's test report for three types of 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 Mrads (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 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. The conditions envelop the plant-specific accident profiles by wide margins. Current and voltage loadings of the cable samples were applied during the 30-day steam exposure. FRC does not regard (i) the differences in spray solution composition or (ii) the absence of an analysis of the nuclear radiation ambient or post-LOCA beta radiation dose to be significant deviations.

Reference 44 is a separate test report for moisture absorption test. The insulation and jacket materials of the test specimen are not identified in the report. According to Reference 46, the test specimen is similar in manufacturing process and has the same basic insulation and jacket compounds as the installed cable.

FRC concludes tht this report establishes the environmental qualification of this equipment item according to the requirements of the Guidelines. This conclusion does not include 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. However, the thermal aging exposure and the simulated LOCA exposure are both very severe. 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.

FRC CONCLUSION:

This equipment belongs in NRC Category II.a. However, FRC is not aware of any proven method of simulating and accelerating all of the factors which affect time-dependent degradation (aging) of non-metallic materials. The Licensee should establish a conservative qualified life and perform the surveillance tests necessary to monitor performance and identify any degradation which would require maintenance or replacement.

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

Equipment items in this category will be fully acceptable and will satisfy all applicable criteria defined in the DOR Guidelines provided that specific modifications are made on or before the designated date. When the modifications are completed, the equipment can be considered qualified. For Big Rock Point Plant there are no items in the category.

4.3.3 NRC Category II.c
EQUIPMENT FOR WHICH DEVIATIONS FROM THE DOR GUIDELINES ARE JUDGED ACCEPTABLE

This section includes equipment items which do not satisfy one or more of the applicable criteria defined in the DOR Guidelines; however, either sufficient bases have been presented to allow a determination that the specific deviations are judged to be acceptable for less than plant life, or the specific deviations are judged by FRC to be acceptable for less than plant life based on review of other applicable qualification documentation associated with the general equipment environmental qualification program.

4.3.3.1 Equipment Item No. 14
Motorized Valve Actuator Located in Turbine Pipe Tunnel

Actuates Backup to MSIV (MO-7067)
(Licensee References 6, 7, 12, and 43)

ORIGINAL FINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE STATEMENT:

This valve is one of several valves that provide backup isolation for the MSIV. No credit for these valves, other than the MSIV, has ever been taken in any safety analysis. The valve is located outside containment in the turbine building pipe tunnel area. It will not be required to operate in the event of an HELB outside containment. The valve is closed on a containment isolation signal on high containment pressure and low reactor water level. Assuming the break is small and operator action is required to close the valve, 10 minutes are assumed to be the time for the operator to actuate the isolation valve switch. The emergency procedure immediate action is to perform this operation. Following closure of the valve, it will remain closed for the duration of the post-LOCA period.

Tests by more than encompass the post-LOCA outside containment environment this valve will be required to operate in. states in the family of actuators is generic in that all actuators use the same materials and the only difference is the size of the actuator. Franklin Test Report No. F-C3441 of a actuator, with Class RH insulated motor, subjected it to a temperature of 340°F and pressure of 108 psig in saturated steam conditions. The unit was pre-aged for 100 hours at 180°C, which by the 10°C rule, is 48 years at 60°C (140°F). The unit was also pre-irradiated to 204 megarads. This test report provides the basis for qualification of the actuator. The actuator motor, a Class B insulated dc-operated, has not been type tested in the actuator assembly.

The outside containment conditions will not be as harsh as those of the tested actuator. The motor, in fact, will only be subject to its normal high-temperature condition in the time until it is required to operate. Radiation at 0.16 Mrads for its 40-year (0.14 Mrads) plus LOCA (20 krads) is within the threshold limit for motor materials listed in the Nuclear Engineering Handbook, H. Etherington, et al. tests of a dc Class H insulated motor and ac Class B insulated motor, in Test Reports B0009 and B0003, irradiated the assemblies to 20 and 10 Mrads, respectively. The dc motor was also aged at 180°C for 100 hours.

Since the effects of thermal and radiation aging on this specific dc motor are undetermined, the motor will be replaced or age qualified by June 30, 1982. The SMB actuator is determined to be qualified. As stated earlier, no credit is taken for this valve in any safety analysis; therefore, no immediate action is required.

FRC EVALUATION:

Although the cited test report references do envelop the Licensee-stated environmental service conditions, the Licensee has not provided evidence

(e.g., a letter from the manufacturer) stating which of these references are applicable for this equipment item. FRC expects that the manufacturer is likely to affirm that valid qualification exists. FRC does not agree with the Licensee's notation on the equipment data sheets, under "Aging," that a qualified life of 40 years has been established, as is discussed in Section 4.1. FRC does agree that qualification tests performed on an actuator are applicable for other sizes, [7] but only when the manufacturer confirms that the cited test report applies.

FRC CONCLUSION:

This equipment belongs in NRC Category II.c based on FRC's review of all known qualification documentation for this equipment. The Licensee has shown that the equipment is likely to function adequately. The Licensee should (i) obtain written confirmation of the applicability of the cited references, (ii) review the aging data and make a conservative estimate of qualified life, and (iii) review the approach used to develop the environmental service conditions for qualification to ensure that this conforms to the requirements of the Guidelines.

4.4 NRC Category III
EQUIPMENT THAT IS EXEMPT FROM QUALIFICATION

The following equipment items are exempt from qualification on the basis that (i) the equipment does not provide a safety function (i.e., should not have been included in the equipment list submitted by the Licensee), (ii) the specific safety-related function of the component can be accomplished by some other designated component which is fully qualified, or (iii) the equipment is not subject to a potentially adverse environment as a result of accidents for which proper operation of the equipment is required. In addition, any failure of the exempt component must not degrade the ability of a qualified component to perform its required safety-related function.

4.4.1 Equipment Item No. 36
Junction Boxes Located Inside Containment
Manufacturer and Model Not Stated
(Original Licensee References 4-2, 4-4, and 4-26)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee submittals list junction boxes located within containment as an item of equipment. FRC considers that the equipment item should consist of the junction box together with internal components (terminal blocks, splices, etc.). References 4-2, 4-4, and 4-26 do not provide qualification test documentation for the junction boxes. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be susceptible to

significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

It is suggested that the Licensee determine from the appropriate manufacturer whether qualification testing has been performed on this item. Consideration should be given to specific testing using junction box enclosures similar to those installed in the plant.

LICENSEE RESPONSE:

Component Junction Boxes JB-160, 161, 164, 166, 167, 170, 180; 170A, B; 171; 171 A, B; IG11A, B, C, D.

In the LOCA environment, the junction boxes provide protection from containment water spray in addition to their normal function of keeping dirt and dust out of the terminal connection. In 1975, Consumers Power conducted an inspection of all junction boxes inside containment. At that time, it was determined some may not withstand the pressure attained during a LOCA. To equalize the pressure differential, all boxes were provided with, or judged to have, an adequate vent to the containment atmosphere. Where boxes were determined to have the possibility to fill with condensate, a drain hole was provided in the bottom of the box. These holes may serve the dual purposes of both providing a vent and/or drain opening. The junction boxes may or may not be equipped with watertight covers. However, it is not necessary to be watertight because of the existence of the drain holes.

Terminal blocks, splice connections and other types of terminal connections may be found in these junction boxes. These connections are discussed elsewhere in this report.

With the existence of drain and/or vent holes, the junction boxes are acceptable.

FRC EVALUATION:

FRC agrees with the Licensee actions to ensure that the inside of the junction boxes are vented to containment to preclude distortion or collapse from LOCA pressure.

FRC CONCLUSION:

This item belongs in NRC Category III because it is not an item of electrical equipment requiring qualification in acccordance with DOR Guidelines.

4.4.2 Equipment Item No. 59
Motorized Valve Actuators Located Within Containment

(Original Licensee References 4-2 and 4-12; Final Licensee References 92.4 and 59)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

Reference 4-1: a test report for two motorized valve actuators and two by another manufacturer. One actuator from each manufacturer was dc powered, and the other ac powered. All four test units have been removed from the Big Rock Nuclear Power Plant and decontaminated prior to the test.

FRC has reviewed this reference and notes the following:

a. The Guidelines require that the test specimen be the same as the equipment being qualified. The installed and tested units are the same since the test units were originally installed at Big Rock Point.

b. The test reported was quite comprehensive and the test unit's equipment items need be qualified for LOCA conditions only. The temperature and pressure profiles used in the test program adequately envelop the plant-specific LOCA profiles.

c. The actuator was not subjected to a documented pre-aging prior to the steam/spray exposure. The Guidelines state that thermal aging of a test specimen is required unless the unit contains no materials susceptible to significant degradation due to aging. The materials used in this equipment have not been identified and evaluated, nor has the natural aging that occurred during the installed life of the tested units at Big Rock Point plant been evaluated. The qualified life has not been established, nor has a program been initiated to monitor failure data for indications of aging-related degradation, as is required by the Guidelines.

d. The actuator was not exposed to nuclear radiation to simulate LOCA conditions, nor was information submitted to demonstrate that the materials used would not be degraded by exposure to nuclear radiations, as is required by the Guidelines.

The Licensee Reference 92.4 states that the Equipment Item 11c valve would only be required for the first 10 minutes after LOCA and that the Equipment Item 59 has been disabled in the closed position and is not required for the accident condition. Accordingly, based on review of Reference 4-12 and other test reports on Limitorque operators, the units are judged to be satisfactory subject to NRC agreement that Item 11c is only required for 10 minutes and that Item 59 is not required to operate after an accident.

LICENSEE RESPONSE:

[None; Equipment Item is not included in Reference 90.]

FRC EVALUATION:

The Licensee noted, in Reference 92.4, that this equipment item has been disabled in the open position.

FRC CONCLUSION:

This equipment belongs in NRC Category III because it does not perform any safety function.

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 WHICH HAS QUALIFICATION TESTING SCHEDULED BUT NOT COMPLETED

The qualification of equipment items in this category has been judged deficient or inadequate based upon FRC's 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. Specific justification for interim plant operation prior to testing should be provided for each item. For the Big Rock Point plant, there are no items in 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 following equipment items is deficient or inconclusive based upon FRC's 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 has not been made available for review. Specific justification for interim plant operation should be provided for each item.

4.5.2.1 Equipment Item No. 2
Electrical Penetration Located Inside Containment
Model Not Stated
(Original Licensee References 4-22 and 4-27;
Final Licensee References 3, 5, 60, and 84)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee stated that letter Reference 4-22 and summary report establishes qualification of the penetrations. FRC has reviewed the referenced documentation and has the following comments and conclusions:

a. Reference 4-22 is a memorandum which discusses methods used by a subcontractor to qualify electrical equipment. It is not a qualification report.

b. Reference 4-27 is a Conax summary report of tests conducted on penetrations for VEPCO's North Anna Station and a certification from Conax that the materials, design, and process used in manufacture of the penetrations for Big Rock Point were the same as for the penetration tested.

c.

Based on the review, FRC concludes that this equipment satisfies the basic requirements of the Guidelines.

LICENSEE RESPONSE:

Component Electrical Penetrations (Type 10 Thru 12)

Type 10: Electrical Penetration with one (1) feedthrough module for 19/C #10 AWG conductors used in control circuits and one (1) feedthrough module for 20/C #16 AWG D-TP used in instrumentation circuits.

Type 11: Electrical Penetration with two (2) feedthrough module used in 19/C #10 AWG conductors for control circuits and one (1) feedthrough module for 20/C #16 AWG D-TP used in instrumentation circuits.

Type 12: Electrical Penetration with two (2) feedthrough modules for 19/C #10 AWG conductors used in control circuits; two (2) feedthrough modules for 20/C #16 AWG D-TP used in instrumentation circuits and one (1) 13/C #12 AWG used in instrumentation circuits.

The subject electrical penetrations are used for Class 1E as well as Nonclass 1E circuits inside the containment. This qualification report, therefore, includes both the functional and the pressure boundary integrity of these penetrations.

1.

The subject electrical penetrations are comprised of a seal body, a conductor feedthrough subassembly, an MK compression ferrule, MK compression cap, terminal blocks, enclosures at each and heat shrink polyolefin tubing.

The seal bodies are fabricated from stainless steel and provide a seal housing for the feedthrough assembly. The feedthrough assemblies are composed of insulated solid conductors, resilient thermoplastic sealants.

The complete assembly is designed to interface with the penetration nozzles which are part of the containment vessel barrier. The nozzle is a Schedule 40, 8" (nominal) OD seamless carbon steel pipe, 12" in length, welded to reactor building shell. The penetration assemblies are attached to the containment vessel nozzle by welding. Each penetration assembly is equipped with two terminal boxes, one on each end.

2. The qualification information of the subject electrical penetrations is based on the following documents and discussions.

a. Document Reference 3 includes Qualification Report IPS-389 for penetrations used in Dresden Nuclear Plant which are similar to the penetrations used at Big Rock Point Plant. The similarity between the electrical penetrations, Types 10 thru 12, and the penetration assembly specimens covered by above report is based on the following facts:

(i) Both of these types of penetrations are designed by Corporation.

(ii) Similar configurations of feedthrough modules (20/C #16 AWG, 19/C #10 AWG and 13/C #12 AWG) have been used in both of these penetrations.

(iii) Conductors of feedthrough modules used in both types of penetrations are insulated with similar insulating materials such as

(iv) Corporation, in Document Reference 2, has certified that production procedures used for the assembly of both types of penetrations are exactly the same.

(v) Corporation, in Document Reference 2, has also certified that similar types of insulating materials have been used for the assembly of both types of penetrations.

During environmental qualification testing covered by Report IPS-389, the penetration assembly, except for terminal block enclosures at both the ends, were first irradiated to 2 x 10^8 rads and then subjected to steam exposure test (to envelop temperature profile given in Figure A1 of IEEE-323, 1974) at 340°F, 60 psig (74.7 psia) and 100% relative humidity. At the end of this LOCA test which continued for 30 days, the electrical penetration assemblies successfully withstood the dielectric and insulation resistance tests.

b. Document Reference 60 is a letter from of Corporation which confirms the similarity between the electrical penetrations used at Dresden Nuclear Plant to the Conax electrical penetrations used at Big Rock Point Plant.

c. Document Reference 84 is a Corporation Report IPS-325 which concludes that various insulating materials like polysulfone, kapton, polyolefin with nuclear grade adhesive (WCSF)-N), etc., are qualified for more than 40-years' life.

d. Document Reference 5 is a Qualification Report IPS-107 for similar electrical terminations subjected to Design Basis Accident Environment for North Anna Power Stations I & II. The similarity between the electrical terminations (for penetration Types 10 thru 12) to the specimen terminations for North Anna penetrations is based on the following facts.

(i) Both of these types of terminations are designed by as a part of complete penetration assembly.

(ii) in Document Reference 2 has certified that production procedures used for the assembly of terminations in both cases are the same.

(iii) in Document Reference 2 has also certified that insulating materials used for the assembly terminations in both cases are the same.

During environmental qualification testing covered by Report IPS-107, the terminations (which include the terminal block enclosures used at both ends of the penetration) were irradiated to 2.5×10^7 rads and then subjected to steam exposure test at 290°F, 57.5 psig (72.2 psia) and 100% relative humidity. At the end of this LOCA test which continued for 10 days, the terminations successfully withstood the dielectric and insulation resistance tests. Analysis of containment parameters shows that following a LOCA, temperature and pressure conditions return to ambient in about 3 days. Therefore, it is concluded that LOCA test duration of 10 days covers the entire time until the conditions in the containment return to essentially ambient.

The Report IPS-107 also indicates that one enclosure containing the terminations was subjected to thermal aging at 300°F for 74 hours which conservatively qualifies it to 22 years' life. Because of the fact that these penetrations were installed after 14 years of the start of commercial operation of the Big Rock Point Plant, it is concluded that the enclosures at both ends of the penetrations qualify for the entire 40-years' life of the plant.

FRC EVALUATION:

The Licensee submittal has been reviewed and compared to the information provided in the November 30, 1978 submittal. FRC has the following comments based upon this review:

a. The testing reported in the references (IPS-389 and IPS-107) envelops the LOCA conditions for the Big Rock Point plant.

b. FRC notes that the previous EEQ submittal (November 30, 1978) letter dated February 15, 1978 stated that the penetrations supplied for Big Rock were the same as those furnished for North Anna plant and provided test report IPS-73.4, dated May 13, 1975, as evidence of qualification. The referenced two letters discussed in the Licensee submittal make no statement concerning production procedures as stated in 2.2.4.b above but merely state that the design basis for the Big Rock penetrations are basically the same. The letter further suggests that a plant-specific report be prepared for the Big Rock plant.

c. With regard to aging and qualified life FRC is not aware of any proven method for simulating and accelerating all of the factors which affect time-dependent degradation of non-metallic materials (aging). While the Arrhenius thermal aging demonstrates that the penetrations should have a long lifetime, it does not unequivocally demonstrate or confirm a lifetime equal to that of the plant.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because identity between the installed units and those tested has not been satisfactorily demonstrated. The manufacturer has cited two different references 5 years apart on apparently different items, based on the description in the reports. FRC would expect that the penetrations used at Big Rock would be qualifiable and suggests that the Licensee follow the manufacturer's recommendations.

4.5.2.2 Equipment Item No. 8
Level Switches Located Inside Containment

Monitors Reactor Water Level for Reactor Scram, Core Spray Valve Opening, and Containment Isolation
(Original Licensee References 4-2, 4-14, 4-16, and 4-18; Final Licensee References 10, 13, 39, 67, 68, 92.4, 15)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee has provided several references relating to the Yarway level indicating switch which monitors reactor water level. These references basically describe a test conducted in 1975 to demonstrate environmental qualification; however, this testing was done on a different model than that which is installed at the Big Rock Plant. Also, a silicone sealant was used for a waterproofing compound for exposed electrical terminals. A hole was drilled in the bottom of the switch enclosure to prevent the enclosure's collapsing under external pressure; mercury switches were used, and electrical connections for the lead entering the switch enclosure were waterproofed. The Licensee did not attempt to account for the differences in the model numbers or for the special features afforded the prototype test unit in relation to the actual units currently in operation at Big Rock Point. The DOR Guidelines state that the testing should be conducted on the same model number or an evaluation should be performed to explain why the differences would not be significant.

FRC's review of the documentation leads to the conclusion that the level switches may be subjected to submergence conditions, and even though the Licensee has stated in Reference 4 that the switch's function of opening a motor operated core spray system valve will only be needed for one minute of the postulated accident, there is no analysis or accident scenario performed

that would be able to provide reasonable assurance that the instrument would not be needed to provide a safety-related function for a longer term incident. In addition, the design of the system's electrical circuit which initiates the core spray valve opening is one in which the level switch contact must open in order to perform its vital function. Since this switch has a high likelihood that water would create electrical tracking across the switch and thereby not allow the motor-operated valve to go to its open position, FRC concludes that having an unqualified switch performing this function is unacceptable.

Reference 18 provides the Licensee's analysis of type testing results conducted in 1975 and offers recommendations as to how the level switch can be altered in order to provide a more qualified instrument to survive the containment's service conditions. Their recommendations included the addition of silicone sealant on terminal block connections; the replacement of the indicator dial, diffuser, and case with Lexan materials (since the indicator had become distorted during the type test and prevented the operation of one of the level switch contacts); the replacement of the plastic plugs with metal plugs; and the changing of the mercury switches with an Acro switch in the instruments. The Licensee has not provided any positive indication of what modifications may have been made or if any follow-up testing was conducted after any of these revisions to the instrument. FRC concludes that the Licensee should provide evidence as to what type of modifications have been performed and how they have resulted in assuring that the unit is environmentally qualified.

The Licensee did not address the DOR Guidelines requirement that materials subjected to thermal or radiation degradation be tested or evaluated. Also, there were no statements regarding qualified life of the instrument. A total material listing was not provided; this may have been able to remove concerns about the instrument's use of elastomers or phenolics that previous testing has shown to be susceptible to thermal stresses.

FRC concludes that the Licensee should provide additional qualification documentation to establish long-term environmental credibility since several materials in the level switch such as the terminal blocks and non-metal components may have significant degradation occuring at elevated temperatures.

The tests conducted on the were limited because they did not address aging, submergence, external pressure, chemical spray, or radiation. The temperature-humidity profile which was employed in the test was not of sufficient duration to envelop the Big Rock Point plant containment profile condition.

In summary FRC concludes the following:

a. 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. The similarity reviewed by FRC thus far does raise additional concerns regarding operability of this manufacturer's level switches.

b. The test specimen was not subjected to thermal aging. The Guidelines state that thermal aging of test specimens is required if the component contains materials known to be susceptible to significant degradation due to aging. The materials used in this equipment have not been analyzed for susceptibility to aging degradation. Therefore, the test result cannot be considered conclusive.

c. The test specimen was not exposed to nuclear radiation to simulate DBE conditions, nor was information submitted to demonstrate that the materials used would not be degraded by exposure to nuclear radiation, as is required by the Guidelines.

d. The level switch has the possibility of being subjected to submergence conditions; however, no testing was performed, nor was a suggestion made by the Licensee to enclose the units in waterproof enclosures.

e. Modifications which may have been made to the unit have not been positively identified and any qualification upgrading cannot therefore be credited.

The documentation submitted to date must be regarded as inconclusive and the Licensee should provide additional documents to establish qualification.

LICENSEE RESPONSE:

Component Level Switches LS-RE09A through H

The LS-RE09A through D level switches are used for three purposes: (1) to provide a reactor scram on low reactor water level, (2) to provide a containment isolation signal on low reactor water level and (3) to provide an open signal to the primary core spray valves MO-7051 and MO-7061. LS-RE09E through H are used to provide an open signal to back up core spray valves MO-7070 and MO-7071 on low reactor water level. The time required for these switches was established at one hour. The basis for this time is that it is the longest time before the low reactor water level set point is reached due to a small break (0.008 ft^2) LOCA. In addition, the level switches must operate in conjunction with PS-IG11A through H to open the core spray valves. The PS-IG11A through H

instruments close contacts in the core spray valve electrical scheme when reactor pressure reaches $\leq$ 200 psig which are in series with the LS-RE09A through H contacts. For the 0.008 ft^2 break LOCA, the reactor depressurizing system (RDS) must first operate to blow the primary system down in order to reach $\leq$ 200 psig. In this scenario, the RDS will actuate when the low reactor water level set point (the same set point as for LS-RE09) is reached via LT-3180 through LT-3183. It should be noted, however, that the reactor operator is directed by procedure to manually initiate the RDS if he deems it necessary. This being the case, it is almost a certainty that the operator will actuate RDS at some time before one hour has elapsed, thus fulfilling the requirements to open the core spray valves. The point being made here is that in all probability the LS-RE09 will have to function in a shorter time and endure a less severe environment than that shown on the qualification sheet. For larger break size LOCA, the time required for LS-RE09 to function automatically is reduced less than two minutes.

The Yarway LS-RE09A through H level switches were modified in 1975 subsequent to a test performed on a similar Model 4416C by the NUS Coproration for Consumers Power (Reference 1). These modifications included (1) removal of the plastic dial cover, dial and diffuser plate, (2) replacement of mercury switches with Acro switches, (3) removal of the light power socket, (4) replacement of plastic plugs on the top of the unit with gasketed metal plugs, and (5) installation of a gasketed aluminum cover (References 67 and 68). The reasons for these modifications are as follows:

1. After exposing the test instrument for approximately three and one-half hours to a saturated steam environment with the temperature and pressure at 275°F and 29 psig, the low-level switch failed to trip because the instrument pointer became bound on the dial/light diffuser that had weakened and sagged due to the prolonged high temperature.

2. The switches were replaced with Acro switches due to seismic considerations.

3. The light power socket was removed in order to avoid subjecting the instrument case to a pressure which could cause it to collapse.

4. The plastic plugs were changed to metal gasketed plugs due to temperature considerations.

5. The gasketed aluminum cover was installed for protection of the open face of the instrument.

In October 5, 1978, it was discovered that the Yarway model 4416C was tested with mercury switches instead of the Acro switches (Reference 10). Corrective action taken by Consumers Power was to send an Acro switch in stock that was procured for the LS-RE09 to the Consumer Power Company System Protection and Laboratory Services Department for a

four-hour test with the temperature and pressure at 132°C (269°F) and 29 psig with 100% relative humidity. The switch was operated hourly. The switch successfully passed this test (Reference 5).

Although it is true that the level switch tested by NUS is not identical to the level switches installed at Big Rock Point, the manufacturer has stated that the only differences between the Model 4416C and the 4420C were in the front and rear housing materials (i.e., naval brass versus bronze) and the switches (i.e., mercury versus Acro). It is concluded, therefore, that the test reports listed as References 13 and 39 are acceptable documentation to qualify pressure and humidity conditions that they will be subjected to during their required function time in the event of a LOCA.

The containment spray water at Big Rock Point is taken from Lake Michigan. It contains no chemicals and is relatively pure water. The LS-RE09 are located in the fuel pit heat exhanger room and, thus, will not be subjected directly to the spray.

The instrument supplier was contacted to provide a list of organic materials used in the 4400 series. The list provided includes:

1. A diaphragm made of Dacron fabric and EPT elastomer.
2. An O-ring on the range adjustment screw made of BUNA N.
3. A backing plate seal made of ethylene-propylene terpolymer.
4. A dust plug made of neoprene.
5. A light diffuser, cover and dial made of Plexiglas and white vinyl (these are no longer installed in LS-RS09A through H).
6. Casing gasket made of cork and rubber compound DK-149.
7. Switch spacers made of laminated melamine.
8. Acro switch made of molded diallyl phthalate.
9. Insulating tubing made of No. 10 clear MIL-I-631C.
10. Terminal blocks made of molded phenolic base.
11. A disc at the bottom of the coil bobbin made of synthane Grade LE.
12. The coil bobbin made of nylon.
13. Sealer made of epoxy.
14. Insulator 0.010" fish paper.

The radiation dose at the containment centerline after two hours is 2 x 10^5 rads. As mentioned earlier, these instruments are unlikely to be required for even one hour. In addition, the instruments are mounted on a substantial concrete wall that would allow one to reduce the dose by a factor of 2. Lastly, although the radiation dose listed assumes 100% core melt, in reality these instruments will perform their functions prior to core uncovery; thus, the actual core damage will probably be no more than that allowed by Technical Specifications for normal operation. According to the Nuclear Engineering Handbook by H. Etherington, et al., of the materials commonly used in electrical equipment, motors, valves, relays, switches, transformers, etc., the only materials that cannot successfully withstand a dose of 2 x 10^5 rads are Teflon and thiskol (polysulfide rubber). As these materials are not in those listed for these instruments, they are considered to be qualified for radiation up to and including the time required for operation.

The instruments have not been tested for thermal aging. Referring back to the list of organic materials, the only material expected to be age sensitive is the BUNA N O-ring on the range adjustment screw. From the drawing, this O-ring does not appear to serve an important function.

In summary, it is concluded from the above discussion that the LS-RE09A through H reactor water level switches will adequately perform their intended function during the LOCA event.

FRC EVALUATION:

The Licensee has provided an equipment analysis and associated test reports as evidence that the level switch is qualified for the pressure, temperature, and radiation environmental service conditions for a time period of 1 hour following a postulated LOCA or SLB in the containment. The service condition of potential submergence were not addressed. Its elevation is listed as 0.5 feet above the estimated flood level, and wave action is a concern.

The Licensee did not provide evidence to address the following concerns that were identified in the DITER:

a. Electrical circuit analysis to review fault in electrical contact or terminal block.

b. Use of silicone sealant on terminal board connections. The radiation component analysis provided by the Licensee did not identify all of the constituent's radiation threshold values for a complete comparison with the radiation environmental service condition of approximately 0.1 Mrad for a 1-hour post-accident gamma dosage. The

Licensee's position that this dosage rate is conservative because of the presence of a shield wall is probably correct; however, a detailed calculation to demonstrate specific dosage has not been provided. Furthermore, the Licensee's logic to deduce that there are no materials with thresholds less than 0.2 Mrad based on the switch components not being listed in a reference handbook is unsatisfactory evidence of radiation qualification.

FRC CONCLUSION:

These level switches belong in NRC Category IV.b because complete documentation associated with radiation degradation and potential flooding wave action has not been addressed with sufficient details. There is a high likelihood of this switch being qualified, but more information on radiation, potential partial submergence, and qualified life is needed.

4.5.2.3 Equipment Item No. 10
Level Transmitters Located Inside Containment

Reactor Water Level (LT-3180 through LT-3183)
(Original Licensee Reference 4-31; Final Licensee References 34 and 41)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The DOR Guidelines state that the test sequence should have the radiation dose either prior to or concurrent with the exposure to elevated temperature and pressure steam environment when the equipment contains items known to be susceptible to radiation degradation. Because the transmitter did not appear to suffer any significant damage from the high radiation dose of 200 Mrads, it does not appear to be sensitive to radiation-related failure and the sequence does not represent a problem.

In conclusion, unless the Licensee can provide other evidence that the transmitter's error and in-leakage problems have been corrected, the transmitter does not have sufficient documentation to demonstrate that it is environmentally qualified.

LICENSEE RESPONSE:

Component LT-3180-3187

The level transmitters are level transmitters which were tested by Franklin Institute Research Laboratories. The test specimen was the same as that installed with the following exceptions: (1) the test specimen had a -400 to 400 inch WC Type 316 stainless steel diaphragm where the installed transmitter had a 0-60 inch WC Type 316 stainless steel diaphragm, (2) the test specimen had a DC 500 silicone oil fill (1,500 psig) vs DC 550 silicone oil fill (3,300 psig) for the installed transmitter, and (3) the test specimen had no "Cleaning for Nuclear Service" where the installed transmitter did. Since the materials are the same and only their ratings are different, the test specimen and that installed are considered essentially identical.

The test sequence was as follows: Steam was added to the test chamber pressurizing it to 58 psig in 9.3 seconds and 68 psig, 304°F in 49 seconds. These conditions were maintained for 3 minutes and then reduced to 288°F, 59 psig and held for 24 hours. The spray solution was turned on at 56 seconds at a rate of 1 gpm. The solution was 1.5% by weight boric acid buffered with NaOH to a pH of 10. After the test, the samples were irradiated at 1 Mrad/h for 192.6 hours and 0.2 Mrad/h for 38.5 hours for a total of 2×10^8 rad. These test conditions are more harsh than those required for qualification, and therefore the guidelines concerning the enveloping of the peaks are met. The test duration is 24 hours. The

temperature profile of the accident condition shows that at 24 hours the temperature is below 120°F. The test was shorter than the temperature/pressure envelope required; however, the peak test temperatures and pressures far exceeded the envelope. Thus, the test duration is felt to be adequate to show qualification of these instruments.

Age related information could not be located. The transmitters are in a normal environment of 50 to 90°F (seasonal variation). CPCo will continue to search for information concerning the existence of age-sensitive materials or aging qualification. Continued operation is justified by the arguments presented in Section 4.

The test duration is considered to be of sufficient duration, and the test conditions exceed those required to demonstrate qualification.

FRC EVALUATION:

FRC notes that the Licensee has not responded to the following specific concerns expressed in the DITER:

- The device exhibited a during the first hours of the simulation test.
-

FRC additionally notes that the Licensee has stated that these transmitters are required for 30 days after the postulated accident and that the instruments are located 6 inches above the flood level.

With respect to test duration, FRC notes that, as stated in the referenced test report, the test chamber time-dependent temperature/pressure profile exceeded the postulated accident profile for but did not totally envelop the required environmental service conditions. The referenced test time duration, did not envelop the required accident profile 41-hour time interval.

FRC CONCLUSION:

This equipment item belongs in NRC Category IV.b because the qualification documentation is deficient with respect to test duration.

However, there is a high likelihood of operability of this device because of the comprehensive testing conducted on this transmitter. The required test duration is 41 hours; the actual test duration was hours. FRC concludes that the Licensee has not justified that transmitter error exhibited during the test is acceptable. FRC also concludes that there is a high probability of in-leakage of water into the instrument, due to its location, which could cause either failure or erroneous readings. The Licensee should address the issue of in-leakage of water in this device.

4.5.2.4 Equipment Item Nos. 11A and 11C
Motorized Valve Actuators Located Within Containment
11A: Class B dc Motor
Actuates CS Valves (MO-7051, -7061)
11C: Class B dc Motor
Actuates MSIV (MO-7050)
(Original Licensee References 4-2, 4-12, and 4-15; Final References 6 and 32)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Guidelines require that, in order to determine the adequacy of the qualification of equipment, the service environment to which the equipment is exposed during normal and accident conditions must be specified. Qualification by type testing requires that the simulated environment of the test envelop the service conditions to be encountered under DBE.

FRC has reviewed the information provided by the Licensee in References 4-2, 4-12, and 4-15 and has concluded that the environmental conditions which would be encountered under a LOCA are enveloped by the test conditions of Reference 4-12 except for aging and radiation exposure. Reference 4-15 presents Consumers Power Co.'s evaluation of the effects of radiation on various materials in the motor operators and states that radiation should not affect the ability of the motor operator to perform its function. Aging is not discussed in the submittal.

FRC has the following comments and conclusions regarding the Licensee submittals:

Tests conducted on Limitorque operators in late 1968 and 1969 included thermal aging.

It is suggested that the manufacturer be contacted to determine whether thermal and radiation test data is available which would apply to the installed valves. Further, it should be determined whether design and material changes made as the result of qualification tests should be applicable to the installed valves.

LICENSEE RESPONSE:

These valve actuators are all dc operated. The valve motors are Peerless with Class B insulation. The actuators are Type A LOCA qualification test for these valves was conducted by Franklin Laboratories for Consumer Power Company in April 1975. Report F-C4124 (Reference 32) describes the test conditions. Qualification of the three subject valves was based on the testing of a similar unit from the BRP Plant. The unit tested was the same actuator size as MO-7051 and -7061 and all are from the same SMA family. The unit also contained a Class B insulated dc motor as do both of the core spray motors. MO-7050 has a larger SMA actuator, but contains the same materials as the actuator tested. The FIRL test exposed the operators for 50 minutes to 240°F and 43.7 psia with tap water spray. Because of the location of the valves, containment sprays will not impinge upon them. A near 100% relative humidity (RH) was maintained throughout most, if not all, of the test by the following means: (1) use of saturated steam to obtain the initial temperature and pressure rise to 240°F, (2) use of fine mist sprays over the specimens and (3) use of saturated steam ejections to maintain chamber temperatures.

Another actuator was tested in a 36-hour steam/28-hour water spray exposure. This particular actuator was equipped with an ac motor. Both satisfactorily passed the test.

The MSIV (MO-7050) closes on a containment isolation signal, either containment high pressure or low reactor water level, and will not be required to operate after closing. Closing of the valve will be accomplished within ten minutes. The core spray valves (MO-7051 and MO-7061) open within one hour on a low reactor water signal and low reactor pressure interlock. Procedurally, after recirculation has begun (maximum of 21 hours), the core spray valves may be closed if the break has been reasoned to be in this core spray line. Closing the valves would be done in conjunction with keeping the backup (redundant) core

spray line in operation to provide core cooling. After closing, it would not be necessary to change the valve lineup.

Radiation and thermal aging qualification testing for the particular actuator and motor has not been done to our knowledge and was not done on the tested actuator. Test Report B0009 [6] for a dc Class H insulated motor aged the unit at 180°C for 100 hours and 2,000 operating cycles and subjected the operator to 10 Mrads of gamma radiation. This was followed by a 25-hour LOCA simulation. There is, of course, no basis to assume the Test Report B0009 qualifies the installed valves by similarity except for both motors operating by direct current.

By a general idea of the materials composing the actuator and motor, they would be able to withstand radiation doses of at least 4.0 Mrads based on Nuclear Engineering Handbook, H. Etherington, et al. Depending on the susceptible materials, they would experience degradation of their physical properties with a dose in excess of 4 Mrads. The normal environmental dose rate is 10 R/h during power operation for the core spray valves. Twenty years at 10 R/h plus a shutdown dose rate of 350 mR/h 60% of the time yields a current dose of 1.4 Mrads. With a conservatively chosen damage value of 4 Mrads, the valve actuators are determined to operate in the radiation environment they experience in a post-LOCA environment.

Thermal aging of the actuator would be accelerated in the event of a LOCA or MSLB inside containment. Such an effect after a 40-year life has not been simulated by type testing. The valve tested in the 1975 test, Report F-C4124, after thirteen years of actual inservice life with normal temperatures ranging from 50°F to 90°F while shut down and 170°F to 200°F when operating, remained operable during the 50-minute LOCA simulation. The normal temperature the core spray valves and MSIV are in is 50°F to 90°F while in shutdown and 100°F to 140°F while the plant is in operation.

Based on the testing, Report F-C4124, and because the test valve had been thermally aged at an accelerated rate during its service life and yet passed the 50-minute LOCA simulation, these valves are considered acceptable for present use. However, no thermal and radiation aging data is available for these valve actuators. Because the aging phenomenon is not a well-known entity and due to the LOCA simulation testing not encompassing the full LOCA envelope, these actuators and motors will either be replaced or rebuilt and requalified by June 30, 1982.

FRC EVALUATION:

Reference 32 is the same as Reference 4-12 and was discussed in the DITER and in the Licensee Response. As the Licensee notes, this test program provides useful -- but not conclusive -- evidence that these MVAs will

function adequately. The Licensee has also provided some analytical reasonings to supplement the deficiencies in the test program. FRC does not believe the analysis of the ability of the materials to withstand nuclear radiation is sufficient to establish qualification. None of the specific materials used in the installed equipment have been identified. FRC also does not agree that a qualified life of 40 years has been established.

FRC believes the decision to rebuild or replace this equipment is unquestionably sound. If the "rebuild" choice is selected, this should include replacement of all nonmetallic components. The Licensee should proceed to establish qualification on a more rigorous basis. Also, the Licensee should make a conservative estimate of the qualified life, and develop a maintenance and equipment performance surveillance schedule to ensure that this equipment will perform properly under accident conditions.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because, although the Licensee has not provided sufficient evidence to show that all the requirements of the Guidelines are satisfied, FRC does believe it likely that this equipment will perform adequately. Because the equipment may have reached, or may be soon reaching, the end of its qualified life, and the qualification documentation is not complete, the Licensee has committed to replacing or rebuilding the installed units. This equipment is acceptaole for interim operation, based in part on systems considerations, provided the installed units are replaced by fully qualified equipment or the units are completely rebuilt prior to July 30, 1982.

4.5.2.5 Equipment Item No. 11B
Motorized Valve Actuators Located Within Containment
Class B ac Motor
Actuates Containment Backup Spray Valve (MO-7068)
(Original Licensee References 4-2, 4-12, and 4-15; Final Licensee Reference 32)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Guidelines require that in order to determine the adequacy of the qualification of equipment, the service environment to which the equipment is exposed during normal and accident conditions must be specified. Qualification by type testing requires that the simulated environment of the test envelop the service conditions to be encountered under DBE.

FRC has reviewed the information provided by the Licensee in References 4-2, 4-12, and 4-15 and has concluded that the environmental conditions which would be encountered under a LOCA are enveloped by the test conditions of Reference 4-12 except for aging and radiation exposure. Reference 4-15 presents Consumers Power Co.'s evaluation of the effects of radiation on various materials in the motor operators and states that radiation should not affect the ability of the motor operator to perform its function. Aging is not discussed in the submittal.

FRC has the following comments and conclusions regarding the Licensee submittals:

> Tests conducted on Limitorque operators in late 1968 and 1969 included thermal aging.

It is suggested that the manufacturer be contacted to determine whether thermal and radiation test data is available which would apply to the installed valves. Further, it should be determined whether design and material changes made as the result of qualifiction tests should be applicable to the installed valves.

LICENSE RESPONSE:

> This actuator contains a with Class B insulation. Presently the valve is not required to operate unless a failure occurs in MO-7064. Operation of both containment sprays would be detrimental to the core spray flow distribution and, therefore, MO-7068 is closed with its breaker pulled. To operate, it would therefore require manual closing of the breaker and manual operation of the controller. If needed, the valve would be used intermittently until the containment post-LOCA environment returned to ambient. From the LOCA envelope, this would take approximately one day. From this period on, the valve will remain closed.

A LOCA simulation test report F-C4124 "Performance Qualification Tests of Four Valve Motor Operators," subjected this valve to a 36-hour test which followed the BRP LOCA envelope for this period of time. The test included 24 continous spraying and intermittent spray for 1-hour periods at hours 27, 31 and 35. Relative humidity was maintained at or near 100% by (1) use of saturated steam to obtain initial temperature and pressure rise to 240°F, (2) use of fine mist water over the test specimen, (3) use of saturated steam injections to maintain chamber temperatures.

Radiation and thermal aging qualification testing has not been done for this type actuator. Generally, most component materials used in the manufacturing of actuators and motors can withstand a threshold damage limit of at least 4.0 Mrads. The effects of thermal and radiation aging are, however, unknown. Based on the LOCA simulation test results though, it is expected this actuator will operate for the required one-day period without significant degradation due to aging. To meet the guideline requirements, the actuator assembly will be replaced or rebuilt and qualified by June 30, 1982.

FRC EVALUATION:

Reference 32 is the same as Reference 4-12, and was discussed in the DITER and in the Licensee Response. As the Licensee notes, this test program provides useful -- but not conclusive -- evidence that these MVAs will function adequately. The Licensee has also provided some analytical reasonings to supplement the deficiencies in the test program. FRC does not believe the analysis of the ability of the materials to withstand nuclear radiation is sufficient to establish qualification. None of the specific materials used in the installed equipment have been identified. FRC also does not agree that a qualified life of 40 years has been established.

FRC believes the decision to rebuild or replace this equipment is unquestionably sound. If the "rebuild" choice is selected, this should include replacement of all nonmetallic components. The Licensee should proceed to establish qualification on a more rigorous basis. Also, the Licensee should make a conservative estimate of the qualified life and develop a maintenance and equipment performance surveillance schedule to ensure that this equipment will perform properly under accident conditions.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because, although the Licensee has not provided sufficient evidence to show that all the requirements of the Guidelines are satisfied, FRC does believe it likely that this equipment will perform adequately. Because the equipment may have reached, or may be soon reaching, the end of its qualified life, and because the qualification documentation is not complete, the Licensee has committed to replacing or rebuilding the installed units. This equipment is acceptable for interim operation, based in part on systems considerations, provided the installed units are replaced by fully qualified equipment or the units are completely rebuilt prior to July 30, 1982.

4.5.2.6 Equipment Item No. 17
Pump Motor Located in the Core Spray Room
General Electric Model SK436XJ1A11
Drives Core Spray Pumps
(Licensee Reference 70)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE STATEMENT:

The core spray pumps are located in the core spray pump room which is heavily shielded from containment. The room contains normal environment except during recirculation phase of the LOCA. During this time, the room reaches 152°F in 400 hours and remains at that level throughout the remainder of the accident. The room temperature is obtained from a calculation which uses motor heat loss and recirculation piping heat loss as heat input to the room. The radiation results from the recirculation of radioactive water from the sump, through the core spray heat exchanger and back to the core. All common motor materials can withstand this radiation level. The pumps are started periodically but have never run for any length of time. The manufacturer states that the motors have a life of 10-years' continuous operation. Since the motors are not used except for LOCA and the fact that they are Class B insulation, leads to the conclusion that they have sufficient life. The motors have a 40°C temperature rise (nameplate data). Class B insulation is good to 130°C so that the room ambient can reach 90°C without harm to the motors. This is greater than the temperature which the room will reach. The lubricating oil is from AMOCO and is American Industrial Oil Number ISO-VG #32. The oil is checked each year and changed when needed. The oil will be changed this refueling outage (1980) to assure adequate life. The oil will be changed periodically thereafter per manufacturer's instructions. The lubricating oil is considered to be able to withstand

the radiation as the levels are so low. Due to the above, the motors are considered qualified.

FRC EVALUATION:

The core-spray pump reference cited by the Licensee provided nameplate information stating that the motor's insulation was Class B and that a 40°C temperature rise could be expected. A detailed analysis or test data were not provided as evidence of operability for the elevated temperature and radiation environmental service conditions.

Motor testing and/or analysis should be used to determine the following motor components' ability to withstand the harsh environmental service conditions of temperature, humidity, and radiation during a postulated LOCA or HELB inside the containment:

a. bearings and lubrication system
b. motor lead and lead-to-cable splices
c. motor insulation system
d. bearing seals.

These motor components have an important role in the establishment of the motor's qualified life which the Licensee should address. In addition, the Licensee should have maintenance surveillance records reviewed and summarized to determine if any abnormal difficulties or bearing wear problems have been experienced which could become the limiting factor in the qualified life determination.

FRC CONCLUSION:

This core spray pump motor belongs in NRC Category IV.b because it has a high likelihood of performing its safety-related function; however, complete documentation has not been provided by the Licensee. The Licensee should provide an oil lubrication review and analysis of its applicability for the 152°F ambient temperature environment combined with the motor's internal bearing friction heat. Also, an overall motor qualified life statement should be provided after reviewing all of the motor's components, such as the bearings, lubrication, seals, and splices, for its environmental service conditions.

4.5.2.7 Equipment Item Nos. 24A and 24B
Solenoid Valves Located Within Containment
24A:
24B:
Actuates Isolation Valves (SV-4876,-4869-4891)
(Original Licensee References 4-1, 4-2, 4-19, 4-20, 4-52, 4-63, and 4-64; Final Licensee Reference 22)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been preaged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

Reference 4-1 is a compilation of memoranda which cover telephone discussions between CPC and personnel regarding the expected capability of the solenoid valve and the aging characteristics of the parts made of Buna N elastomers. No test results applicable to the installed valves are provided. Reference 4-2 contains an evaluation of the solenoid valves which states that the materials are satisfactory for radiation and temperature conditions of LOCA and that failure of the coil due to humidity or moisture would result in the valve failing in the "safe" direction. Reference 4-19 restates information contained in 4-1 and 4-2. Reference 4-20 recommended that the plastic parts be replaced. Reference 4-52 states that replacement of parts in would be deferred until the January 1979 refueling outage.

Reference 4-63 is an letter recommending replacement of and s ating that they believed the coil would be satisfactory for LOCA conditions. Reference 4-64 is an internal memo which discusses service conditions of various solenoid valves and states that the plastic parts of Model would be changed in the January 1979 refueling outage.

From a review of the referenced correspondence, FRC has the following comments and conclusions:

a. There is no evidence from the Licensee submittals that qualification tests meeting the Guidelines requirements have been conducted on any of the solenoid models identified as Equipment Item Number 1. In addition, FRC has reviewed available reports of tests on solenoid valves and has not found test results which would be applicable to the valves installed in Big Rock Point plant.

b. In tests conducted on solenoid valves, failures have occurred as a result of congealing and hardening of lubricant, causing the valve to stick. Failure of energized Class H insulated coils occurred at environmental temperatures of 290°-300°F, and excessive seat leakage in elastomer seated valves has occurred as a result of irradiation and temperature.

c. has recommended to various utility companies that elastomer parts (Buna-N) be replaced on a periodic basis to preclude adverse effects of aging and irradiation.

d. FRC concludes that the information provided by the Licensee does not establish whether the valves installed at Big Rock Point plant will or will not fail when subjected to the LOCA/HELB accident condition, nor is the mode of failure predictable from the Licensee submittal.

The Licensee should obtain from the manufacturer either evidence of tests on the valves involved or detailed analysis of data from tests of similar valves so that the qualification status of the valves with respect to the DOR Guidelines is established.

LICENSEE RESPONSE:

These solenoid valves are automatically de-energized by the reactor safety system on an isolation trip from either the containment high-pressure or low-reactor water level signals. The valves are supplied with redundant valves outside containment which are also tripped by the same signals. The valves outside containment will not be subject to a harsh environment during a LOCA inside containment except for higher than normal radiation exposure. Once these valves are de-energized by the

isolation trip, they remain in this mode. Immediate operator action in the emergency procedure is to actuate the isolation switch to close these valves. This action would back up the automatic trip or provide a trip in the event one hadn't been initiated. Since the valves are tripped automatically on a high containment pressure signal at 1.5 psig, or manually prior to this, they will not have to operate in as severe an environment as described on the qualification sheet. has stated the peak temperature and pressure of 235°F and 41.7 psia will probably not have any effect on the operability of the valves. From Figures 1 and 2 when the pressure reaches 1.5 psig, temperature is below 140°F. Relative humidity of 100% may cause shorting of the solenoid according to ASCO. In this event, the result will be de-energizing of the solenoid and closing of the control valve. It will fail safe. The valves are located in areas that are protected from the containment sprays but will have operated prior to the spray initiation which has a higher containment pressure set point. The 2-hour radiation exposure of 0.2 Mrads was the dose that could be received at the center line of containment. This dose can be halved as the valves are mounted on large concrete walls. Conservatively using 0.1 Mrads, the valve materials of BUNA N and Zytel 103 have a threshold limit of greater than 1 Mrad, which qualifies the valves to operate with the radiation exposure they receive. The 30-day exposure of radiation to the valves of 0.73 Mrad can also be halved and is also below the threshold limit of the valve materials. The materials will not degrade to failure with the valves in their de-energized mode during the post-LOCA period, due to the radiation exposure.

Both the SV-4869 and 4891 have, within the past month, had replacement spare parts installed in them. The replacement parts include the BUNA N O-ring and seat which is also the elastomer used in SV-4876 along with Zytel 103. SV-4876 is not presently on a PM replacement program. It, however, has a metal seat and Zytel disc with O-rings of BUNA N. Zytel, in addition to having a radiation resistance of 5 Mrads, is also a high temperature material (Reference 22) which has been used in the scram solenoids; and, in one test, endured 204 hours at a maximum temperature of 415°F and 250,000 operations. The material has not been shown to be susceptible to aging failures.

The valves are considered acceptable for present use because of their ability to withstand LOCA temperature, pressure and radiation. Replacement of parts and age-sensitive materials also provides acceptability. There are also redundant valves located outside containment which de-energize on an isolation signal and remain in this mode. According to ASCO, failure of the materials with the valve de-energized will leave the valve in its fail-safe position.

The effects of humidity will also leave the valve in its fail-safe position. The valves, however, do not meet the qualification guidelines as type testing of these valves has not been done. Therefore, the valves will be replaced by June 30, 1982.

FRC EVALUATION:

Reference 22 is an engineering report on tests of the Zytel 103 nylon disc. In one test, the disc was continuously energized for 264 hours, and then was operated 250,000 times. The maximum temperature the disc experienced due to energization was 415°F. The report indicates that the material has not shown any aging failures after the test. Since the installed valves are always in the energized state, the discs in these vales will experience relatively high internal temperatures at all time. Owing to the lack of aging basis, the relationship between the test conditions and the estimated life cannot be established; therefore, the aging effect of Zytel is uncertain. FRC believes, however, that these solenoids are likely to function to close early in the accident scenario, but their capability for subsequent functioning (if required) is uncertain.

The Licensee should make a conservative estimate of the qualified life and institute a program to asertain whether any in-service failures during the installed life of the equipment are the result of aging degradation.

FRC CONCLUSION:

This equipment item belongs in NRC Category IV.b. Although no evidence of qualification has been provided, FRC believes it is likely that the equipment will function satisfactorily because of operating time considerations. The Licensee has committed to replace these valves with qualified equipment by June 30, 1982. In the meantime, the valves should be placed in the preventive maintenance program.

4.5.2.8 Equipment Item No. 25
Solenoid Valve Located Within Containment

Actuates Isolation Valve (SV-4879)
(Original Licensee References 4-1, 4-2, 4-19, 4-20, 4-52, 4-63, and 4-64)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT

The Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification

methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

Reference 4-1 is a compilation of memoranda which cover telephone discussions between CPC and personnel regarding the expected capability of the solenoid valve and the aging characteristics of the parts made of Buna N elastomers. No test results applicable to the installed valves are provided. Reference 4-2 contains an evaluation of the solenoid valves which states that the materials are satisfactory for radiation and temperature conditions of LOCA and that failure of the coil due to humidity or moisture would result in the valve failing in the "safe" direction. Reference 4-19 restates information contained in 4-1 and 4-2. Reference 4-20 recommended that the plastic parts be replaced. Reference 4-52 states that replacement of parts in HT831677 would be deferred until the January 1979 refueling outage. Reference 4-63 is an ASCO letter recommending replacement of and stating that they believed the coil would be satisfactory for LOCA conditions. Reference 4-64 is an internal memo which discusses service conditions of various solenoid valves and states that the plastic parts of Model HT831677 would be changed in the January 1979 refueling outage.

From a review of the referenced correspondence, FRC has the following comments and conclusions:

a. There is no evidence from the Licensee submittals that qualification tests meeting the Guidelines requirements have been conducted on any of the solenoid models identified as Equipment Item Number 1. In addition, FRC has reviewed available reports of tests on

solenoid valves and has not found test results which would be applicable to the valves installed in Big Rock Point plant.

b. In tests conducted on solenoid valves, failures have occurred as a result of congealing and hardening of lubricant, causing the valve to stick. Failure of energized Class H insulated coils occurred at environmental temperatures of 290°-300°F, and excessive seat leakage in elastomer seated valves has occurred as a result of irradiation and temperature.

c. has recommended to various utility companies that elastomer parts (Buna-N) be replaced on a periodic basis to preclude adverse effects of aging and irradiation.

d. FRC concludes that the information provided by the Licensee does not establish whether the valves installed at Big Rock Point plant will or will not fail when subjected to the LOCA/HELB accident condition, nor is the mode of failure predictable from the Licensee submittal.

The Licensee should obtain from the manufacturer either evidence of test on the valves involved or detailed analysis of data from tests of similar valves so that the qualification status of the valves with respect to the DOR Guidelines is established.

LICENSEE RESPONSE:

This solenoid valve is normally de-energized with its three redundant resin sluice control valves closed during power operation. When de-energized, the solenoid and control valves are in their fail-safe mode. The control valves are only opened in transferring resins and are opened only for short periods of time. Resin transfer takes place on an average of once per year and complete transfer can be done in about four hours. The valves are closed on an isolation signal from either high containment pressure or low reactor water level. If open, the valve will be closed automatically within a maximum of one hour following a break. See Section II.D. For small breaks, manual action by an operator is assumed in 10 minutes. Automatic initiation of an isolation for large breaks will occur within seconds of a break. Immediate operator action following symptoms of a break is to actuate the isolation valve switch which will back up or initiate valve closure. A normally closed manual valve outside containment also provides a redundant isolation valve. In the improbable event a break occurs when the line is open, the automatic and redundant valves inside containment provide the containment boundary.

The pilot head in the solenoid is the same model used for the scram solenoid valves and was installed in 1978. It has a NEMA 4 watertight enclosure that will prevent the 100% humid atmosphere from affecting the performance of the valve. Since the automatic isolation signal will be

initiated at 1.5 psig, or isolation may be manual prior to this, the valves will not have to operate in as severe an environment as described on the qualification sheet. From Figures 1 and 2 when pressure reaches 1.5 psig for all break sizes, temperature is less than 140°F. A temperature of 235°F and a pressure of 41.7, has stated, will not affect the ability to operate the solenoid. The valve elastomers are normally in ambient conditions inside containment whereas the scram solenoids of the same valve model normally are energized and are at a substantially higher temperature.

Aging deterioration from the containment temperature will be accelerated in the 3-day post-LOCA period in which the atmosphere returns to ambient. The operation of the solenoid will not be affected, however. With the present 5-year preventative maintenance program, the elastomers will remain qualified with respect to thermal aging.

The 30-day radiation dose of 0.73 Mrads for the radiation dose in air listed on the qualifiction report sheet can be halved due to the solenoid valve being mounted adjacent to a substantial cement wall. The BUNA N, eoprene and Zytel 103 materials all have radiation damage threshold values greater than 1.0 Mrads according to and other sources. Radiation aging will not be significant with the dose received and the preventative maintenance program will provide new components to maintain the qualification.

The solenoid valve is mounted in a metal cabinet which will protect it from any containment spray water.

The valve is normally de-energized; and, if energized, will de-energize prior to the atmosphere becoming hostile, and will not be subject to any spray. It has a watertight enclosure which will prevent humidity from causing any shorting. The accident 30-day radiation dose is below threshold damage to its materials. It also has a periodic replacement program to replace aging susceptible materials. A manual valve outside containment also provides backup to the normally closed control valves. Based on these reasons, the valve, although not meeting guideline requirements, is considered acceptable as is.

FRC EVALUATION:

Because the seals and other components of the valve may be degraded by the normal service environment, and because a high temperature steam environment may exist for several minutes before functioning (i.e., change of position) is called for, the Guidelines now require that a qualification test be performed for a minimum of 1 hour under LOCA conditions to verify proper

operation. The effects of the normal service environment on the equipment should be taken into consideration and the qualified life explicitly determined.

FRC CONCLUSION:

This equipment item belongs in NRC Category IV.b. Although no evidence of qualification has been provided, FRC believes it is likely that the equipment will function satisfactorily. The Licensee has stated that the equipment is presently under the five-year preventive maintenance program.

4.5.2.a Equipment Item No. 26
Solenoid Valve Located Within Containment

Actuates Isolation Valve (SV-4892)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE STATEMENT:

This solenoid valve isolates the treated waste to containment via manual operation from the control room. The control valve is normally closed with the solenoid de-energized during the power operation. The control valve is in series with a check valve. Both valves are leak rate tested each refueling outage to verify the containment integrity. Manual operation would only be necessary if the control valve were opened to transfer water to the spent fuel pool system. In this event the check valve provides the containment boundary until the control valve is closed. The Plant Emergency Procedure will have to be modified to identify to the operator to close the control valve. After identification of a LOCA, closure of the valve would be expected within 10 minutes, which is the allowed operator action time. The action will certainly be done in the one-hour minimum operating time requirement given in the NRC guidelines.

The BUNA N materials used in the nonmetallic valve components will not be significantly affected by the LOCA temperature of 235°F during the time it takes to operate the valve. has stated this material is acceptable for continuous operation in ambient temperatures up to 180°F and up to 240°F for short periods of time. The LOCA pressure of 41.7 psia also should not affect the function of the valve. Similar solenoid valve enclosures routinely are subjected to higher pressures during containment integrated leak rate tests. has also stated that 100% relative humidity may cause the valve to short. This, however, will

result in the valve de-energizing and closing the control valve. The valve is located in an area that is protected from the containment sprays and is mounted adjacent to a substantial concrete wall that will have the effect of halving the radiation dose. In the time it would take to operate the valve, less than 0.1 Mrads (2-hour gamma dose halved) would be absorbed. BUNA N has a threshold limit of 1.0 Mrads, which is well above the 30-day gamma dose of 0.365 Mrads which would be conservatively assumed. Thermal and radiation aging does affect the BUNA N material; however, the valve is presently in a preventative maintenance replacement program to replace the aging-susceptible components.

Although no qualification testing has been done on this model valve, the length of time required, insignificant effect of pressure and temperature on the operation, low radiation dose, and because it is backed up by another valve which will fail passively, all provide reasons for its present acceptability. Furthermore, it is a normally closed valve. Failure due to shorting or aging deterioration, in conversations with ASCO, will result in a fail-safe operation; i.e., the control valve will shut. Because the guidelines are not met, an acceptable replacement will be procured and installed by June 30, 1982.

FRC EVALUATION:

Because the seals and other components of the valve may be degraded by the normal service environment and because a high-temperature steam environment may exist for several minutes before functioning (i.e., change of position) is called for, the Guidelines require that a qualification test be performed for a minimum of 1 hour under LOCA conditions to verify proper operation. The effects of the normal service environment on the equipment should be taken into consideration and the qualified life explicitly determined.

FRC CONCLUSION:

This equipment item belongs in NRC Category IV.b. Although no evidence of qualification has been provided, FRC believes it is likely that the equipment will function satisfactorily because the time of active functioning occurs early in the accident scenario. FRC notes that the Licensee has committed to replacement prior to June 30, 1982.

4.5.2.10 Equipment Item No. 29
Solenoid Valves Located in the Turbine Building

Actuates Backup MSIVs (SV-4899, -4916)
(Licensee Reference 49)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

NONE

LICENSEE STATEMENT:

Both of these solenoids pilot control valves which are backup valves to the main stream isolation valve. They are not required to operate in an HELB outside containment. The valves will be required to be manually operated as they do not receive a safety system initiation. There also has never been any credit taken for the operation of these valves in any safety analysis. Reliance has been placed solely on the MSIV. Immediate operator action after he has determined the symptoms of a break are to close the automatic isolation valves via a hand switch. This will either back up or initiate their closure. Since these valves are not automatically actuated, the procedure will have to be modified such that their common hand switch is also actuated by the operator. Operation action is assumed as 10 minutes. The valves must, therefore, remain operable for this period of time. For the 30-day period, the valves will remain de-energized with the control valves shut.

The normal and post-LOCA environments with respect to pressure and humidity are at normal, nonharsh conditions. Higher than normal temperatures in the area near the solenoids are, when the plant is at power, the result of the nearby main steam line. These temperatures will remain high through the time the valves are required to close but will decay to ambient shortly thereafter. Both valves are normally energized when the plant is at power. When de-energized, the solenoid vents the air from the control valve, closing it. In fail-safe mode, according to A failure of the BUNA N seats will not result in actuation of the control valve.

The radiation dose accumulated for a 40-year life is approximately 0.14 Mrads. In the 10 minutes until the time to operate the valve, there will be no significant additional exposure. The 30-day, post-LOCA integrated dose is 20 krads, which results in a total lifetime exposure of 0.16 Mrads. This falls within the radiation 25% damage limit of 7.6 Mrads, based on compression set for the BUNA N elastomers used in the valve.

BUNA N is known to be susceptible to the effects of thermal aging; it shows a good resistance to radiation. As discussed earlier, the temperatures that the valves are exposed to during power operation are above normal. Presently, the valves are not on a preventative maintenance part replacement program to replace the age-susceptible

parts. This will be done, however, by June 30, 1982, and be continued on a five-year program. The failure mode of the valves due to deterioration of the BUNA N seats would be leakage through the seats resulting again in a fail-safe operation of the control valve.

Based on the fail-safe failure mode of the valve and because no credit in the safety analysis has been taken for these valves, they remain acceptable as is. However, to maintain correct operability of the valve, it will be maintained on a preventative maintenance schedule initiated by June 30, 1982.

FRC EVALUATION:

Because the seals and other components of the valve may be degraded by the normal service environment, and because of a main steam line nearby causing the environment to have higher than normal temperatures, the Guidelines require that a qualification test be performed for a minimum of 1 hour under LOCA conditions to verify proper operation. The effects of the normal service environment on the equipment should be taken into consideration and the qualified life explicitly determined. The preventive maintenance (PM) program should commence immediately.

FRC CONCLUSION:

This equipment item belongs in NRC Category IV.b. Although no evidence of qualification has been provided, FRC believes it is likely that the equipment will function satisfactorily. It is recommended that the PM program for this equipment begin immediately.

4.5.2.11 Equipment Item No. 30
Solenoid Valves Located Within Containment

Actuates Valves (SV-4980 thru 4983)
(Original Licensee References 4-1, 4-2, 4-19, 4-20, 4-52, 4-63, and 4-64)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records, reflecting a comprehensive qualification methodology, be referenced for review for all Class 1E equipment. Type

testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

Reference 4-1 is a compilation of memoranda which cover telephone discussions between CPC and personnel regarding the expected capability of the solenoid valve and the aging characteristics of the parts made of Buna N elastomers. No test results applicable to the installed valves are provided. Reference 4-2 contains an evaluation of the solenoid valves which states that the materials are satisfactory for radiation and temperature conditions of LOCA; failure of the coil due to humidity or moisture would result in the valve failing in the "safe" direction. Reference 4-19 restates information contained in 4-1 and 4-2. Reference 4-20 recommended that the plastic parts be replaced. Reference 4-52 states that replacement of parts in HT831677 would be deferred to the January 1979 refueling outage.

Reference 4-64 is an internal memo which discusses service conditions of various solenoid valves and states that the plastic parts of model HT831677 would be changed in the January 1979 refueling outage.

From a review of the referenced correspondence, FRC has the following comments and conclusions:

a. There is no evidence from the Licensee submittals that qualification tests meeting the Guidelines requirements have been conducted on any of the solenoid models identified as Equipment Item Number 1. In addition, FRC has reviewed available reports of tests on solenoid

valves and has not found test results which would be applicable to the valves installed in Big Rock Point plant.

b. In tests conducted on solenoid valves, failures have occurred as a result of congealing and hardening of lubricant causing the valve to stick. Failure of energized Class H insulated coils occurred at environmental temperatures of 290°-300°F, and excessive seat leakage in elastomer-seated valves has occurred as a result of irradiation and temperature.

c. ASCO has recommended to various utility companies that elastomer parts (Buna-N) be replaced on a periodic basis to preclude adverse effects of aging and irradiation.

d. FRC concludes that the information provided by the Licensee does not establish whether the valves installed at Big Rock Point plant will or will not fail when subjected to the LOCA/HELB accident condition, nor is the mode of failure predictable from the Licensee submittal.

The Licensee should obtain from the manufacturer either evidence of test on the valves involved or detailed analysis of data from tests of similar valves so that the qualification status of the valves with respect to the DOR Guidelines is established.

LICENSEE RESPONSE:

These solenoid valves are pilot valves for the reactor depressurization system (RDS) isolation valves. The valves are actuated on RDS initiation signals from low steam drum level, fire pump start (core spray water), low reactor water level and a two-minute time delay. The solenoids are normally de-energized and are energized on RDS initiation. The RDS function is for small to intermediate size breaks to relieve reactor pressure so that the low-pressure core spray system can provide core cooling. Once open, the isolation valves shall remain open; however, for most all breaks requiring RDS, the reactor becomes depressurized and there will be no increase of pressure in the primary system above the core spray injection pressure; i.e., pressure will be relieved out the break and, therefore, the RDS would not be required except for the initial depressurization. Only for a very small break would it be necessary for the valves to remain operable and only one of the four valve trains would then be required.

The valves have been recently modified to replace the Celcon plastic disc holder with a metal disc holder. The solenoid is equipped with a Class H high-temperature coil. Class H coils are also used in ASCO's NP series solenoids that have passed type testing to meet the IEEE-323 (1974) requirements. Test Report AQS21678/TR, Rev A. The coils are

suitable for ambient temperatures up to 212°F for continuous operation. Temperature is above 212°F for only about 1000 seconds for the spectrum of breaks shown in Figure 1 of Section II. The coil is also radiation resistant (the test irradiated the coils to 200 Mrads). The coil is also suitable in high humidity conditions. The containment sprays are not likely to impinge on the valves as they are located above and behind the nozzles. The watertight enclosures are suitable to keep both humidity and water spray, which is not caustic, out of the coil area. The explosionproof enclosure will also withstand the LOCA pressure of 41.7 psia.

The BUNA N valve elastomers have a radiation resistance of 7.0 Mrads based on 25% degradation due to compression set. This is a factor of 10 larger than its lifetime plus LOCA dose. The coils, as stated above, have been tested to 200 Mrads.

The valves were installed in the plant in 1976 and by June 30, 1982 will have not aged significantly as they are normally de-energized except for testing and are in a nonharsh, normal environment. Due to lack of qualification type testing, they will be replaced by the June 30, 1982 date. Presently, they are acceptable because of their materials construction to withstand the six-year life plus a 30-day post-LOCA period.

FRC EVALUATION:

Because the seals and other components of the valve may be degraded by the normal service environment, and because a high temperature steam environment may exist for several minutes before functioning (i.e., change of position) is called for, the Guidelines require that a qualification test be performed for a minimum of 1 hour under LOCA conditions to verify proper operation. The effects of the normal service environment on the equipment should be taken into consideration and the qualified life explicitly determined. As is noted in Section 2.2.4, Supplement 2 to IE Bulletin 79-01B requires that qualification be demonstrated. (This requirement was established subsequent to the preparation of FRC's DITER.)

FRC CONCLUSION:

This equipment item belongs in NRC Category IV.b. Although no evidence of qualification has been provided, FRC believes it is likely that the equipment will function satisfactorily.

4.5.2.12 Equipment Item No. 31
Solenoid Valves Located Within Containment

Actuates Reactor Depressurization Valves (SV-4984 Through 4987)
(Original Licensee Reference 4-8; Final Licensee Reference 23)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Guidelines require that, in order to determine the adequacy of the qualification of equipment, the service environment to which the equipment is exposed during normal and accident conditions be specified. Qualification by type testing requires that the simulated environment in the test chamber envelop the service conditions to be encountered under DBE.

FRC has reviewed test report information contained in Reference 4-8 and concludes that the environmental conditions which would be encountered under LOCA are enveloped by the test conditions except for chemical spray. The Guidelines define water as chemical or water spray. Since the Big Rock Point plant has containment spray, the effects of spray should have been evaluated. The Licensee should establish with the valve supplier whether there is data on the effects of chemical or water spray and provide evidence that chemical or water spray will not adversely affect the performance of the solenoid valve.

LICENSEE RESPONSE:

These solenoid valves are pilot valves for the reactor depressurization valves. The valves were installed in 1976. The valves are actuated by RDS initiation signals from low steam drum level, fire pump start (core spray water), low reactor water level and a two-minute time delay. On RDS initiation, the valves are energized. The function of RDS is to depressurize the primary system so that the low-pressure core spray system can provide core cooling. Once energized, the valves will remain energized throughout the post-LOCA period; however, for most all breaks requiring RDS, the reactor becomes depressurized and there will be no increase of pressure in the primary system above the core spray injection pressure; i.e., pressure will be relieved out the break and, therefore, the RDS would not be required except for the initial depressurization. Only for a very small break would it be necessary for the valves to remain operable and only one of the four valve trains would then be required.

The valves are mounted above and behind the containment spray nozzle distribution pattern and are, therefore, not subject to spray. Big Rock Point's containment sprays are, at any rate, water from Lake Michigan and

are not corrosive and would not be expected to be detrimental to the valve operation. Furthermore, the solenoids have NEMA Type 4 waterproof enclosures.

Reference 23 describes testing of solenoid operated valves. The test specimens were production model valves of the similar design using the same materials and solenoid coil type as was used in the Big Rock Point model valves. The testing also included the NEMA Type 4 junction box with terminal board which is the same as on the installed valves. The unit was pre-irradiated to 33 megarads, aging was conducted as a temperature-humidity wear-out test and the valve was heated by passing fluid through it to 200 ± 10°F with the test chamber air temperature at 150 ± 10°F and relative humidity of 55 ± 5%. The valve was then cycled times. The chamber conditions were maintained for hours. The accident simulation test temperature and pressure increased then to 340°F within where it was held for and then reduced to ambient in This was then repeated. Relative humidity was maintained at 100% and demineralized water spray was used. After the repeat, the test chamber was maintained at The pressure was then reduced to ambient and temperature maintained at for

Although the preaging did not simulate a 40-year life, the LOCA simulation temperatures being significantly above the Big Rock Point break envelope have enough margin to provide additional aging data of the test assembly. Big Rock Point's maximum temperature of 235°F lasts for only a few seconds, temperatures above 200°F for the large break (Figure 1) are only a 1000-second period and, lastly, atmospheric temperatures are reached within one day. 200°F for 30 days by the 10°C rule provides about 3-years' simulated life at 104°F.

The valve elastomers are ethylene-propylene rubber, silicone rubber and an asbestos and rubber insulator washer. The asbestos and rubber washer is captured above and below the solenoid coil. Although it may deteriorate in its life, it will remain intact due to its position. The EPR and silicone rubber O-rings are also captured and their failure will not affect the operation of the valve.

Based on the testing of the solenoid valve with the same NEMA Type 4 enclosure, Class H coil and nonmetallic materials, the valves are considered acceptable for their remaining lifetime. Three of the four pilot solenoids are scheduled for replacement in the upcoming refueling outage.

The aging of the materials will not be detrimental to the operation of the solenoid and, at normal ambient temperatures with the solenoid normally de-energized, the materials are not subjected to an accelerated aging process.

FRC EVALUATION:

FRC has reviewed Reference 23, a test report of solenoid operated valve, and has the following comments:

a. The position switch failed upon exposure at and during the accident simulation. The time at which the failure occurred was not stated. Examination after the test revealed that the entire switch housing was filled with water, due to the improper installation of piping to the NEMA Type 4 enclosure. The Licensee did not state whether the installation in the plant followed the manufacturer's recommendation.

b. Pre-aging of equipment is not adequately addressed and the qualified life is not established. Aging should be performed prior to the DBE simulation. The time duration at the elevated temperature during the DBE simulation cannot be credited as part of the thermal aging test and used for the estimation of the qualified life. Therefore, the Licensee's statement with regard to the simulated life at 104°F is invalid.

c. The model of the test specimen is not the same as the installed equipment. The Licensee states that the design and the materials used are similar to the one installed.

d. The steam exposure adequately enveloped the plant-specific profile.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because, although there is no evidence of qualification, the equipment is likely to perform its intended function. It is recommended that the Licensee determine the maintenance status of these items to assure that there is a reasonable amount of life remaining.

4.5.2.13 Equipment Item No. 35
Solenoid Valves Located Within Containment

Actuates Valves
(Original Licensee References 4-1, 4-2, 4-19, 4-20, 4-52, 4-63, and 4-64; Final Licensee Reference 50)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records, reflecting a comprehensive qualification

methodology, be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been pre-aged may be considered acceptable provided the component does not contain materials which are known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

Reference 4-1 is a compilation of memoranda which cover telephone discussions between CPC and personnel regarding the expected capability of the solenoid valve and the aging characteristics of the parts made of Buna N elastomers. No test results applicable to the installed valves are provided. Reference 4-2 contains an evaluation of the solenoid valves which states that the materials are satisfactory for radiation and temperature conditions of LOCA and that failure of the coil due to humidity or moisture would result in the valve failing in the "safe" direction. Reference 4-19 restates information contained in 4-1 and 4-2. Reference 4-20 recommended that the plastic parts be replaced. Reference 4-52 states that replacement of parts in HT831677 would be deferred to the January 1979 refueling outage.

Reference 4-64 is an internal memo which discusses service conditions of various solenoid valves and states that the plastic parts of model HT831677 would be changed in the January 1979 refueling outage.

From a review of the referenced correspondence FRC, has the following comments and conclusions:

a. There is no evidence from the Licensee submittals that qualification tests meeting the Guidelines requirements have been conducted on any of the solenoid models identified as Equipment Item Number 1. In addition, FRC has reviewed available reports of tests on solenoid valves and have not found test results which would be applicable to the valves installed in Big Rock Point plant.

b. In tests conducted on solenoid valves, failures have occurred as a result of congealing and hardening of lubricant causing the valve to stick. Failure of energized Class H insulated coils occurred at environmental temperatures of 290°-300°F, and excessive seat leakage in elastomer-seated valves has occurred as a result of irradiation and temperature.

c. has recommended to various utility companies that elastomer parts (Buna-N) be replaced on a periodic basis to preclude adverse effects of aging and irradiation.

d. FRC concludes that the information provided by the Licensee does not establish whether the valves installed at Big Rock Point plant will or will not fail when subjected to the LOCA/HELB accident condition, nor is the mode of failure predictable from the Licensee submittal.

The Licensee should obtain from the manufacturer either evidence of test on the valves involved or detailed analysis of data from tests of similar valves so that the qualification status of the valves with respect to the DOR Guidelines is established.

LICENSEE RESPONSE:

Component Solenoid Valves SVNC-22A, B, C, D and SVNC-27A2A Through SVNC-27F5B

These solenoid valves, the master scram, the scram valve solenoids and scram dump tank isolation valve solenoids operate automatically on a reactor protection signal or by manual action. In the event of a small break LOCA where the operator observes symptoms of a LOCA and takes manual action to scram the reactor, the operation time is considered to be ten minutes. In the event of a larger break LOCA, the scram will be automatic and within seconds after the break. Following the scram, it is inconsequential if the valves fail in the LOCA environment.

These valves have had replacement spare parts kits installed in 1979. A thermal aging problem has been determined to exist due to the solenoids being constantly energized resulting in the components being subjected to a constant heat source. The elastomers, BUNA N, neoprene and Zytel 103, are the materials replaced in the five-year preventative maintenance program. Being constantly energized, the solenoid enclosure skin temperature has been measured to be approximately 125°F. Internal temperatures within the core range above 200°F.

Based on the data provided by General Electric in Reference 50, this model valve can operate in sustained periods of up to seven years without failure in the energized mode.

For automatically initiated scrams, these solenoids will be actuated by the safety system before the temperature or pressure has risen much above ambient; e.g., at three seconds (Figures 1 and 2) for the large break, the temperature is 150°F and the pressure is 21 psia. The internal valve materials will not be affected by the LOCA temperature. If operator action is required to initiate a scram in event of a LOCA, the break would be small and both temperature and pressure would not have risen much above ambient. The result is that these valves will be exposed to a less severe environment than noted on the qualification sheet. Furthermore, has stated the valve materials can withstand the peak LOCA temperatures. These are not, of course, significantly larger than the internal valve temperatures. ASCO has also stated the solenoid enclosures are of sufficient strength to withstand the LOCA pressure of 41.7 psia without affecting the operation of the valve. The solenoid has a NEMA 4 watertight enclosure and, therefore, 100% relative humidity will also not be a factor in the operation of the valve.

A two-hour gamma radiation dose of 2.0×10^5 rads in air is conservatively used for the integrated 10-minute dose. The actual dose would, of course, be much less and, in fact, be neglible since, in any event, scram will occur long before the core is uncovered. The valve nonmetallic materials all have a threshold damage limit greater than the two-hour integrated dose. Radiation will, therefore, not affect the valve operation.

Aging effects on the valve will not be significantly accelerated in the LOCA event, but do occur as stated above due to the continuous energization of the solenoid. The valves have been placed on a five-year program to replace the nonmetallic components. The five-year program was established in response to IE Bulletin 78-14. Based on the replacement program, the valves are considered qualified for a 40-year plus LOCA lifetime.

The valves become submerged at some time after the accident. Submergence, however, will not occur prior to a scram. After a LOCA and scram have occurred, the position of the control valves becomes inconsequential; therefore, submergence also becomes unimportant.

In summary, the valves are considered qualified based on their 10-minute operating time and component replacement program which will maintain the valve qualification. Even though they do not meet guideline requirements for equipment inside containment, they are acceptable as is.

FRC EVALUATION:

Because it is constantly energized, the solenoid valve experiences high internal temperature causing the nonmetallic materials of construction to

degrade. The manufacturer has recommended periodic inspection and replacement of coils and elastomeric materials (3-5 years). The Licensee has established a five-year preventive maintenance program to replace the nonmetallic materials. FRC believes that these solenoid valves are likely to function to close early in the accident, but their capability for subsequent functioning (if required) is uncertain.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because, although there is no evidence of qualification, the required safety function is likely to be completed before the environment becomes "harsh," and subsequent functioning is not required. The Licensee has committed to a five-year replacement program.

4.5.2.14 Equipment Item No. 37
Splice Insulation Located Inside and Outside Containment
3M, Model Not Stated
(Licensee References 24 and 82)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

The splices connecting the penetration pigtails to the cables were covered with waterproofing tape as part of the fire protection modification performed in 1978. The splice now only provides the mechanical and electrical connection between the two cables. The waterproofing tape provides the qualification.

The tape used in the test and in the plant was Scotch Brand 23 (ethylene-propylene material) wrapped first with Scotch Brand 33+ (a vinyl material) wrapped on top. To assure proper adhesion, the cables were first prepared using the kit specified by 3M. The preparation and taping was done in accordance with strict procedure.

The waterproofing tape was environmentally tested. The test specimens were spliced conductors whose splices were covered with waterproof tape and were placed in the test chamber. A dielectric test was performed on each spliced cable and then the cable was loaded to 6 amps. Steam was introduced into the test chamber and the conditions of 235°F and 41.7 psia were reached in 10 seconds. The temperature was held for 1 hour. The conditions were allowed to decay (held to a linear decay) to 115°F

and 16.9 psia in a 24-hour period. These conditions were held for four days. While the chamber was sealed, a dielectric test was repeated. The specimens were removed from the chamber and sprayed for 20 minutes with water from a 1-1/2" fire hose with a head pressure of 80 psig. The dielectric tests were then repeated. The spliced cables passed all of the tests.

The radiation resistance of the tape is as follows: at 5×10^7 R, dielectric was essentially the same and elongation at break was approximately 1/2 of the values at zero radiation.

No age-related data could be obtained. The splices and tape are in an environment of 50°F to 90°F (seasonal variation).

The splices, which are covered with waterproof tape, are considered qualified because of the following: The tape was environmentally tested to a temperature-pressure envelope that matched the required envelope for the first hour and exceeded the required envelope for the next 19 hours. The test duration was not as long as required; however, the test exceeded the envelope through most of the test and at the time of which the test was terminated, the containment would have been less than 125°F. The spray test was much more severe and the degradation due to radiation was acceptable.

The effects of thermal aging on these materials will be investigated by June 30, 1982; however, since the materials are relatively new (fall 1978 installation) and since the normal environment averages 70°F, these splices are concluded to be acceptably qualified in the interim.

FRC EVALUATION:

This Licensee submittal, including a reference letter and test report, has been reviewed, and FRC has the following comments:

a. The testing on the splices and potting compound to seal the cables meets the LOCA conditions for Big Rock plant but has no margin.

b. The spray was not simultaneous with the LOCA.

c. Radiation was neither sequential nor simultaneous with LOCA.

d. No pre-aging had been done on the tape or potting.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because, even though the Guidelines test requirements have not been complied with, FRC considers that

there is a good probability of operation. Testing as required by the Guidelines should be performed on a representative sample.

4.5.2.15 Equipment Item No. 42
Electrical Cables Located Inside Containment
Anaconda Model 32277
(Licensee Reference 29)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

600 volt class cables, sizes as detailed below, with EPRC (ethylene propylene rubber) insulation and hypolan (chorosulfonated polyethylene) insulation.

- 1/C #4/0, 19 Strand
- 1/C #1/0, 19 Strand
- 1/C #6 AWG 7 Strand
- 3/C #12 AWG 7 Strand

The above cables are used in Class 1E circuits and are qualified for the severe environments for use inside containment as below:

A. Document Reference 29 demonstrates that the cable is qualified for use inside the containment based on 40-year life and LOCA. The test cycle included the following and meet 1EEE-383 (1974) and 1EEE-323 (1975) requirements:

1. Thermal aging at 150°C (302°F) for 168 hours to simulate a 40-year life.

2. Gama radiation from a Cobalt-60 source for a total dosage of 2 x 10^8 R.

3. LOCA simulation test with chemical spray as below:

- 346°F at 110 Psig for 8 Hours
- 335°F at 96 Psig for 3 Hours
- 315°F at 69 Psig for 4 Hours
- 265°F at 28 Psig for 81 Hours
- 212°F at 4 Psig for 26 Days

The chemical spray has a pH range of 9 to 11 and consisted of 3,000 ppm of boron, 0.064 molar sodium thiosulfate and sodium hydroxide to meet the pH requirements. BRP spray is noncorrosive lake water.

4. Per Document Reference 29, the cable passed dielectric test when immersed in water. The duration of the test is 26 weeks.

5. Per FIRL Test Report No F-C4350-3 attached with Document Reference 29, chemical spray was included in the test cycle. The surface of the cable is almost fully covered with fluid for the duration of the spray and this is generally equivalent to submergence. Coincidental high pressure (113 psig) and temperature (346°F) in the LOCA test chamber give raise to a situation which is analogous to submergence under high pressure and temperature.

FRC EVALUATION:

In addition to the references cited, FRC has reviewed other testing on the installed cable (F-C3341 and F-C3033) which envelop the Licensee-stated conditions for LOCA and MSLB. However, none of the testing subjects the cable to submergence under LOCA conditions and the ICEA tests do not envelop the LOCA submergence.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because qualification has been demonstrated for all conditions except submergence for which testing needs to be done. Furthermore, FRC is not aware of any proven method of similarity and accelerating all of the factors which affect time-dependent degradation (aging). The Licensee should establish a conservative qualified life and perform the surveillance necessary to identify any degradation which would require maintenance or replacement.

4.5.2.16 Equipment Item No. 43
Electrical Cables Located Inside Containment
Model Not Stated
(Licensee References 8, 28-30, and 51)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE·

600 Volt Class Cable 1/C #2 AWG With EPR (Ethylene Propylene Rubber) and Neo (Neoprene) Insulation and Jacketing Materials

The subject cable is used in Class 1E circuits and is qualified for use inside the containment as detailed below:

A. Per Document Reference 51, the cable is qualified for use inside the containment based on 40-year life and LOCA. The test cycle included the following:

1. Thermal aging at 121°C for seven days to simulate 40-year life.

2. Gamma radiation from Cobalt-60 source for a total dosage of 2 x 10^8R.

3. LOCA simulation test with chemical spray as below:

- 346°F at 113 Psig for Hours

-

-

-

-

The chemical spray has a pH range of 9 to 11 consisting of boric acid (3,000 ppm) and sodium hydroxide to meet the pH requirements. BRP containment spray is Lake Michigan water which is noncorrosive.

4. After LOCA test, the specimens passed the dielectric test when immersed in water at 80 V ac/mil for five minutes.

5. Document Reference 29, 8 and 30 prove the generic capability of the cables with ethylene propylene rubber (EPR) insulation to meet the severe environments inside the containment.

6. The cable meets the ICEA (formerly IPCEA) standard requirements including long-term moisture absorption test which is done by immersion in water.

7. During LOCA tests described in Paragraph A.3, chemical spray was included in the test cycle. The surface of the cable is almost fully covered with fluid for the duration of the spray and this is generally equivalent to submergence. Coincidental high pressure (113 psig) and high temperature (346°F) in the LOCA test chamber give test rise to a situation which is analogous to submergence under high pressure and temperature.

It is concluded that the temperature, pressure, humidity, spray, submergence, radiation and aging requirements have been met an exceeded by substantial margins.

FRC EVALUATION:

FRC considers that the report establishes the environmental qualification of this equipment item according to the requirements of the Guidelines except for submergence. 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. However, thermal aging exposure and the simulated LCOA exposure are both very severe. 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. However, none of the findings subjects the cable to submergence under LOCA conditions and the ICEA tests do not envelop LOCA submergence.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because qualification has been demonstrated for all conditions but submergence, which needs to be done. However, FRC is not aware of any proven method of simulating and accelerating all of the factors which affect time-dependent degradation (aging) of non-metallic materials. The Licensee should establish a conservative qualified life and perform the surveillance tests necessary to monitor performance and identify any degradation which would require maintenance or replacement.

4.5.2.1 Equipment Item No. 44
Electrical Cables Located Inside Containment

(Original Licensee References 4-9, 4-10, and 4-11; Final Licensee References 31, 72, and 74)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Guidelines require that the model of the test specimen be the same as that of the equipment being qualified. The type test is valid only if the installed equipment and test specimen have the same design, material and production procedures. An analysis of the impact of deviations between the test specimen and the installed equipment is an essential part of the qualification documentation. In a case when cable model numbers are not usually applicable, however, the manufacturers' formulations and identification of the insulation and jacket materials are considered to be an acceptable alternate to a model or part number. For example, the descriptions provided on sheet number 516726 Paragraph B of Reference 4-9, page 2-1 of Reference 4-10, and page 2 of Reference 4-11 provide adequate identification of the cable.

FRC has reviewed the referenced documentation and notes the following:

a. The Licensee submittals do not identify the installed cables in a manner which permits independent verification that the type cable installed is the same as the type tested. As noted during the site visit, the Licensee should identify the installed cables in such a manner that it can be established

b. Regarding Note 1 of Reference 92.4, the Guidelines state that specifying saturated steam as a service condition during type testing of equipment that will become flooded in service is not an acceptable alternative for actually flooding the equipment during the test. Items subjected to steam and spray need to be tested under steam and spray conditions. Items subjected to submergence need to be tested under conditions of submergence.

It is suggested that the Licensee contact the manufacturers to determine whether there are tests of submergence applicable to the installed cables.

LICENSEE RESPONSE:

600 volt class cables, sizes as detailed below, with Kerite FR insulation and jacketing material.

- 1/C #12 AWG
- 2/C #14 AWG
- 3/C #14 AWG
- 7/C #14 AWG

The above cables are use in Class 1E circuits and are qualified for use inside the containment as below:

A. Document Reference 31 details the actual type testing performed under simulated post-accidednt reactor containment service conditions for the cables with identical insulation and jacket materials as the subject cables. During this test, the cables were subjected to gamma radiation of 1.2×10^8 rads from Cobalt-60 source while being exposed to saturated steam at 82 psig (96.7 psia) and a temperature of 325°F for 13 hours followed by exposure at 5 psig (19.7 psia) at 228°F for 7 days. Borated chemical spray with a pH of 9.5 was sprayed during this test.

   The test specimens were energized during the LOCA test. After the LOCA test, it passed dielectric test (4 kV) when applied for 5 minutes.

   It is concluded that the cable meet or exceed the radiation, temperature, pressure, spray, submergence and humidity requirements as discussed below:

   1. The peak temperature inside the containment of 235°F psia after an accident at Big Rock exists for less than 1 hour and decreases to 80°F in less than 3 days. The tested temperatures and duration exceed the requirements.

   2. The peak pressure inside the containment of 41.7 psia after an accident exists for less than 1 hour and decreases linearly to 15 psia in less than a day. The tested pressure and duration exceed the requirements.

B. Based on the following evaluation, it is concluded that the cables meet the aging requirements.

   1. The cable meets the physical property requirements of ICEA (formerly IPECA) Standards such as tensile strength and elongation after air oven test for 168 hours at 121°C.

2. The radiation aging to a dose of 1.2×10^8 R gamma, far exceeds the requirement of 7.3×10^5 (Gamma) and $1.3 \times 7.3\ 10^7$ beta. The level of beta at the cable insulation surface is decreased by jacket material, conduit and and tray cover, etc.

   From this highter test value for radiation, it is concluded that the cable will withstand higher temperature for longer duration to qualify for aging requirements.

3. The subject cables were installed in 1970 and the aging requirements (when compared to the rest of the equipment in the Plant) is for less than 32 years.

4. The temperatures and durations during LOCA test detailed in Paragraph A above are much higher than the accident values. This supplements the evaluation given above for aging qualification.

C. Qualification for Submergence, Spray

1. The cable meets the ICEA (formerly IPCEA) Standard requirements, including long-term moisture absorption test which is done by immersion in water.

2. During LOCA tests described in Paragraph A, chemical spray was included in the test cycle. The surface of the cable is almost fully covered with fluid for the duration of the spray and this is generally eqivalent to submergence. Coincidental high pressure (82 psig) and high temperatures (325°F) in the LOCA test chamber give rise to a situation which is analogous to submergence under high pressure and temperature.

Summary

1. Paragraph A provides evidence for temperature, pressure, humidity and radiation qualfications.

2. Paragraph B provides an evaluation for aging qualification.

3. Paragraph C provides an evaluation for spray and submergence.

FRC EVALUATION:

FRC concludes the report establishes the environmental qualification of this equipment item according to the requirements of the Guidelines except for submergence. This conclusion does not include 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. However, the thermal aging exposure and the simulated LOCA exposure are both very severe. 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. The LOCA testing does not include submergence. The ICEA test conditions listed by Licensee do not envelop LOCA submergence, temperature, and pressure.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because submergence has not been demonstrated and needs to be done. The Licensee should establish a conservative qualified life and perform the surveillance tests necessary to monitor performance and identify any degradation which would require maintenance or replacement.

4.5.2.18 Equipment Item No. 45
Electrical Cables Located Inside Containment

(Original Licensee References 4-9, 4-10, and 4-11; Final Licensee Reference 35)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Guidelines require that the model of the test specimen be the same as that of the equipment being qualified. The type test is valid only if the installed equipment and test specimen have the same design, materials, and production procedures. An analysis of the impact of deviations between the test specimen and the installed equipment is an essential part of the qualification documentation. In a case when cable model numbers are not usually applicable, however, the manufacturers' formulations and identification of the insulation and jacket materials are considered to be an acceptable alternate to a model or part number. For example, the descriptions provided on sheet number 516726 Paragraph B of Reference 4-9, page 2-1 of Reference 4-10, and page 2 of Reference 4-11 provide adequate identification of the cable.

FRC has reviewed the referenced documentation and notes the following:

a. The Licensee submittals do not identify the installed cables in a manner which permits independent verification that the type cable installed is the same as the type tested. As noted during the site visit, the Licensee should identify the installed cables in such a manner that it can be established

b. Regarding Note 1 of Reference 4, the Guidelines state that specifying saturated steam as a service condition during type testing of equipment that will become flooded in service is not an acceptable alternative for actually flooding the equipment during the test. Items subjected to steam and spray need to be tested under steam and spray conditions. Items subjected to submergence need to be tested under conditions of submergence.

It is suggested that the Licensee contact the manufacturers to determine whether there are tests of submergence applicable to the installed cables.

LICENSEE RESPONSE:

1. 600 Volt Class Cables, Sizes as Detailed Below, With XLPE (Radiation Cross-Linked Polyolefin) Insulation and Jacketing Material

   - 2/C #14 AWG
   - 3/C #14 AWG
   - 3/C #16 AWG
   - 6/C #14 AWG
   - 12/C #16 AWG
   - 2/C #16 AWG Twisted and Shielded
   - 2/C #16 AWG Twisted

2. 1000 Volt Class Cables, Sizes as Detailed Below, With XLPE (Radiation Cross-Linked Polyolefin) Insulation and Jacketing Material

   - 2/C #12 AWG
   - 2/C #10 AWG
   - 2/C #12 AWG
   - 3/C #10 AWG

The subject cables are used in Class 1E circuits inside the containment in areas not subjected to submergence. The qualification is based on the following documents and discussion:

A. Document Reference 35 demonstrates that the cables with identical insulation is qualified for use insid the containment based on

40-year life and LOCA with chemical spray. In containment, environmental conditions at Big Rock Point are less severe (including the absence of chemical spray) than the parameters to which the cable is type tested. BRP sprays are lake water which is not corrosive. The test cycle included the following:

1. Thermal pre-aging at 150°C (302°F) for 25 days and at 160°C (320°F) for 12 days followed by thermal and radiation aging at 150°C (302°F) and 5×10^7 R for 7 days.

2. Combined LOCA and radiation exposure as below:

   - Radiation 1.5×10^8 R.

   - Temperature and pressure as follows:

   177°C (351°F) at 84.7 psia for 10 hours; 135°C (275°F) at 45.7 psia for 4.5 days; 100°C (212°F) at 24.7 psia for 26 days.

   The chemical spray during LOCA consisted of 3,000 ppm of boron and 0.064 molar sodium thiosulfate and adjusted with sodium hydroxide to a pH value of between 9.5 to 11.

3. Dielectric test for five minutes while immersed in water.

B. Raychem Flamtrol cables meet ICEA (formerly IPCEA) standard requirements including long-term moisture absorption test which is done by immersion in water.

During LOCA tests described in Paragraph A.2, chemical spray was included in the test cycle. The surface of the cable is almost fully covered with fluid for the duration of the spray and this is generally equivalent to submergence. Coincidental high pressure (70 psig) and high temperature (351°F) in the LOCA test chamber give rise to a situation which is analogous to submergence under high pressure and temperature.

It is concluded that the temperature, radiation, humidity, spray, submergence, and aging requirements have been exceeded by a substantial margin and, therefore, the cable is qualified for this application.

FRC EVALUATION:

FRC concludes the report establishes the environmental qualification of this equipment item according to the requirements of the Guidelines except for submergence. This conclusion does not include 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. However, the thermal aging exposure and the simulated LOCA exposure are both very severe. 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. The LOCA testing does not include submergence and the ICEA test conditions listed by the Licensee do not envelop LOCA submergence, temperature and pressure. It is noted that the Licensee has not determined from the manufacturer whether the submergence test of Appendix V applies to Big Rock.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because qualification for submergence has not been demonstrated and needs to be done. The Licensee should establish a conservative qualified life and perform the surveillance tests necessary to monitor performance and identify any degradation which would require maintenance or replacement.

4.5.2.19 Equipment Item Nos. 19A and 19B
Pressure Switch Located in the Electrical Penetration Room

19A: 12L-AA5-FSS
19B: 4NN-E411-YX5TT
Senses High Containment Pressure
(Original Licensee Reference 4-44; Final Licensee Reference 44)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications or Post-Incident Equipment Review Sheets are not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been pre-aged may be considered acceptable provided the component

does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

Because this instrument is located in an environment which is only subjected to 100°F and 15 psia ambient conditions, one would expect that its ability to function in this type of environment would not be a concern. The 100% relative humidity, however, could present a problem even though it is anticipated to be of short duration due to a high-energy pipe break. Similarly, radiation may pose an environmental problem for which the switch must be qualified.

FRC concludes that the Licensee should obtain documentation from the manufacturer relative to the humidity and radiation exposure.

LICENSEE RESPONSE:

Component Pressure Switches PS-7064A, B and PS-664 Through PS-667

These switches are located in the electrical penetration room. The room is adjacent to the containment and has a common wall (the shell) with it. PS-664 through -667 are part of the reactor protection system causing a scram on high containment pressure. Once it actuates, the signal is sealed in. PS-7064A and B actuate on high containment pressure also. Actuation of this switch starts the timer in the enclosure spray valve opening circuitry. Once actuated, the signal is sealed in. As a result, the switches will only experience the environment, which transferred through the sphere, due to the containment atmosphere at the high containment trip set point (1.7 psig). The highest containment wall temperature at the trip set point is about 150°F. This results from a very small steam line break. Larger breaks result in faster time of high containment pressure trip and lower wall temperatures. The temperature at the switches due to this break is less than this due to thermal inertia in the containment wall, atmosphere of the cable penetration room and switch housing. Since the temperature inside the switch is so low, it is considered to be qualified for temperature. The switches are enclosed in an explosion-proof housing or waterproof housing and are not affected by humidity. The normal environment in the room is 40°F-100°F (seasonal variation). Since the conditions at time of actuation are not much different from normal, no significant stress will be placed on the switches during the event. Therefore, the switches are likely qualified for aging. For further arguments, see "Aging" in the body of the report.

The radiation levels are based on 100% core melt at t=0 integrated to one hour after the start of the event. For the large breaks LOCAs, the core uncovers early in the event, but high containment pressure also occurs very early (< 1 hour) so that the radiation levels will be less. For

the small main stream line break upon which the 1-hour operational time is based, the core does not uncover and 100% core melt will not occur. Even if forced to assume 100% core melt at t=0 for a small main steam line break, the radiation levels are such that all common materials used in the construction of these switches can withstand them. Therefore, the switches are qualified for radiation.

FRC EVALUATION:

Because the pressure switches are required to be qualified for one hour plus the expected time period for the switch to perform its safety function, the environment in the electrical penetration room should be considered harsh instead of mild. Reference 90, which was submitted by the Licensee, stated that the postulated LOCA occurrence condition would result in an elevated temperature condition of 178°F. The radiation level is expected by the Licensee to be no greater than 0.076 Mrad.

No additional reference documentation was provided by the Licensee to demonstrate environmental qualification.

Sufficient detailed radiation analysis which would compare expected specific radiation exposures with the exposure that would degrade the specific switch's components has not been conducted.

A statement that specifically assesses a qualified life value needs to be provided by the Licensee. Also, the equipment maintenance surveillance records should be reviewed and summarized to determine if any abnormal difficulties have been experienced which could become the limiting factor in the qualified life determination.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b because appropriate qualification documentation has not been made available to demonstrate total compliance with the DOR Guidelines. The Licensee has not analyzed this switch in sufficient detail or addressed future testing options which could demonstrate the switch's ability to withstand the environmental service conditions. Also there was no attempt to identify any anticipated aging degradation that would become a limiting factor in its qualified life.

4.5.2.20 Equipment Item No. 34
Solenoid Valves Located Within Containment

(Original Licensee References 4-1, 4-2, 4-19, 4-20, 4-52, 4-63, and 4-64)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records, reflecting a comprehensive qualification methodology, be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been preaged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

Reference 4-1 is a compilation of memoranda which cover telephone discussions between CPC and personnel regarding the expected capability of the solenoid valve and the aging characteristics of the parts made of Buna N elastomers. No test results applicable to the installed valves are provided. Reference 4-2 contains an evaluation of the solenoid valves which states that the materials are satisfactory for radiation and temperature conditions of LOCA; and failure of the coil due to humidity or moisture would result in the valve failing in the "safe" direction. Reference 4-19 restates information contained in 4-1 and 4-2. Reference 4-20 recommended that the plastic parts be replaced. Reference 4-52 states that replacement of parts in HT831677 would be deferred to the January 1979 refueling outage.

Reference 4-64 is an internal memo which discusses service conditions of various solenoid valves and states that the plastic parts of model would be changed in the January 1979 refueling outage.

From a review of the referenced correspondence, FRC has the following comments and conclusions:

a. There is no evidence from the Licensee submittals that qualification tests meeting the Guidelines requirements have been conducted on any of the solenoid models identified as Equipment Item Number 1. In addition, FRC has reviewed available reports of tests on solenoid valves and has not found test results which would be applicable to the ASCO valves installed in Big Rock Point plant.

b. In tests conducted on solenoid valves, failures have occurred as a result of congealing and hardening of lubricant causing the valve to stick. Failure of energized Class H insulated coils occurred at environmental temperatures of 290°-300°F, and excessive seat leakage in elastomer seated valves has occurred as a result of irradiation and temperature.

c. has recommended to various utility companies that elastomer parts (Buna-N) be replaced on a periodic basis to preclude adverse effects of aging and irradiation.

d. FRC concludes that the information provided by the Licensee does not establish whether the valves installed at Big Rock Point plant will or will not fail when subjected to the LOCA/HELB accident condition, nor is the mode of failure predictable from the Licensee submittal.

The Licensee should obtain from the manufacturer either evidence of test on the valves involved or detailed analysis of data from tests of similar valves so that the qualification status of the valves with respect to the DOR Guidelines is established.

LICENSEE RESPONSE:

Component Solenoid Valve SVNC-22F, G, H, J

The scram dump tank vent valves prevent pressure buildup in the scram dump tank which may, if the control rod drive is isolated, cause withdrawal of a control rod when the dump tank pressure would exceed reactor pressure. The valves are normally energized and, on a scram signal, they de-energize after a one-minute time delay. Once de-energized, the valves remain in this position. A scram following a

break would be initiated either automatically on low drum level or high containment pressure or, in the event of a small break, operator action would initiate a scram. Time for operator action is 10 minutes. The solenoids would have to, because of the time delay, operate in the 235°F temperature, 41.7 psia pressure and 100% relative humidity.

The valve being normally energized has internal core temperatures in excess of 200°F. The BUNA N and Zytel materials are subjected to these internal temperatures. The LOCA environment will not subject the valve materials to higher than normal temperatures during the minute it must remain energized. The LOCA pressure should also not affect the operation of the valve. solenoids are routinely pressurized in containment leak rate testing without affecting the operation of the valves. Relative humidity of 100% will also not affect the watertight NEMA type of enclosure.

The valve, once de-energized, must remain in this state to maintain the dump tank vent path. The valves must then fail-safe and the effects of radiation aging and submergence should not affect them. An integrated gamma dose for two hours in air of 0.2 Mrads plus the beta-gamma dose in water for 30 days of 4.0 Mrads yields 4.2 Mrads, which is within the damage limit of the susceptible materials. states BUNA N is acceptable to 7.0 Mrads and Zytel 103 to 5 Mrads. NRC Guidelines, Table C-1, states neoprene acceptable to 10 Mrads. The most significant aging of the valves is not in the post-LOCA atmosphere, but rather during the normal energizing of the valve. This, in fact, led to a preventative maintenance replacement program to replace the materials on a five-year schedule. The high post-LOCA temperatures, as stated above, will not subject the materials to additional stress during the time it takes for them to operate. has also stated BUNA N is acceptable in continuous temperatures up to 180°F and 240°F for short periods. The materials in the post LOCA atmosphere will not be exposed to temperatures in excess of these. Submergence also should not cause a failure mode which will cause opening of the control valve. With the solenoid de-energized, failure of the diaphragms and seals will only result in fail-safe operation according to

In summary, the valves have been evaluated to be acceptable for post-LOCA operation. They, however, do not meet the guideline requirements and, because of this and the need to maintain the venting of the dump tank, they will be replaced by June 30, 1982.

FRC EVALUATION:

Because the seals and other components of the valve may be degraded by the normal service environment, and because a high temperature steam environment may exist for several minutes before functioning (i.e., change of position) is called for, the Guidelines now require that a qualification test

be performed for a minimum of 1 hour under LOCA conditions to verify proper operation. The effects of the normal service environment on the equipment should be taken into consideration and the qualified life explicitly determined.

FRC CONCLUSION:

This equipment item belongs in NRC Category IV.b. Although no evidence of qualification has been provided, FRC believes it is likely that the equipment will function satisfactorily. The Licensee has stated that the equipment is presently under the five-year preventive maintenance progam.

4.5.2.21 Equipment Item Nos. 12A and 12B
Motorized Valve Actuators Located Within Containment
12A: w/Class B dc Motor
Actuates Containment Valve (MO-7064)
12B: w/Class B ac Motor
Actuates Backup CS Valves (MO-7070, -7071)
(Original Licensee References 4-2, 4-12, and 4-15; Final References 6, 16, 17, 32, and 69)

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

The Guidelines require that, in order to determine the adequacy of the qualification of equipment, the service environment to which the equipment is exposed during normal and accident conditions be specified. Qualification by type testing requires that the simulated environment of the test envelop the service conditions to be encountered under DBE.

FRC has reviewed the information provided by the Licensee in References 4-2, 4-12, and 4-15 and has concluded that the environmental conditions which would be encountered under LOCA are enveloped by the test conditions of Reference 4-12 except for aging and radiation exposure. Reference 4-15 presents Consumers Power Co. evaluation of the effects of radiation on various materials in the motor operators and states that radiation should not affect the ability of the motor operator to perform its function. Aging is not discussed in the submittal.

FRC has the following comments and conclusions regarding the Licensee submittals:

Tests conducted on Limitorque operators in late 1968 and 1969 included thermal aging.

It is suggested that the manufacturer be contacted to determine whether thermal and radiation test data is available which would apply to the installed valves. Further, it should be determined whether design and material changes made as the result of qualification tests should be applicable to the installed valves.

LICENSEE RESPONSE:

Equipment Item No. 12A:

This actuator contains a dc motor with Class B insulation and weatherproof enclosure. The actuator was installed in the plant in 1970. The valve is actuated by containment high pressure and presently has a 15-minute time delay. The valve will be required to actuate intermittently in the post-LOCA environment until the containment atmosphere returns to ambient. From the LOCA pressure-temperature envelope this will be approximately 1 day. Subsequently, the valve will remain closed.

Report F-C4124 "Performance Qualification Tests of Four Valve Motor Operators" subjected an exact type of actuator and motor to a 36-hour LOCA simulation. The simulation followed the LOCA envelope curve for this period of time. The test also included 24 continuous hours of spraying for the first 24 hours and then intermittent 1 hour spraying at hours 27, 31 and 35. Relative humidity was maintained at or near 100% by (1) use of saturated steam to obtain initial temperature and pressure rise to 240°F, (2) use of fine mist water over the test specimen, (3) use of saturated steam ejections to maintain chamber temperatures. Radiation and thermal aging were not included as part of the test procedure. has conducted testing for aging qualification. Technical Report TR422 subjected an NA type actuator with a Class B insulated motor to an aging test which simulated 48 years at 60°C. The unit in the test was placed in a furnace for 100 hours at 180°C. 48 years at 60° was determined by the 10°C rule. TR422 did not include a LOCA simulation, and although an NA type actuator was used in the test, the only difference between the NA and 14A units, according to is the

motor hous ng material, cast iron versus cast aluminum. The results of the test, states, and the electrical characteristics will be exactly the same for the 14A as the NA type units. Test No. N/14/2 used a prototype unit built for material evaluation for the Standard "A" and NA1 components. The actuator to MO-7064 being a Standard "A" type unit. The test results showed the "A" range components capable of withstanding 30 megarads during their 40-year life, according to The test unit, however, was equipped with a Class H insulated motor. actuators (Report B0003) with Class B, ac motors have been irradiated to 20 megarads and been subjected to outside containment HELB conditions for a 16-day period and have operated satisfactorily.

Another Limitorque test (Report B0009) subjected a Class H, dc motor to 10 megarads of gamma radiation and followed with a 25-hour HELB simulation. The unit was also pre-aged at 180°C for 100 hours and satisfactorily passed the test.

The 30-day radiation dose given on the qualification sheet can be divided in half as the actuator is mounted adjacent to a 3.5-foot thick concrete wall. An integrated dose of 0.37 Mrads is within the qualification dose of most materials. A one-day dose of 0.49 Mrads halved, giving 0.25 Mrads is what the actuator will see during the time required to operate. This is also well within almost any material radiation resistance.

In summary, based on the testing of similar type units for radiation and thermal aging and the test conducted by CP Co, the actuator is expected to operate for the required one-day period. After one day, the valve will not be required to operate and will remain closed. The motor starter is installed in the station power room outside containment and will not be subjected to a harsh environment; therefore, a misoperation due to failure of the starter is not a credible failure. The actuator is considered acceptable for use.

Equipment Item No. 12B:

These actuators have ac motors with Class B insulation and weatherproof enclosures. MO-7070 and MO-7071, the backup core spray valves, were initially installed in the Plan in 1970. The core spray valves are actuated on low reactor water level coincident with reactor pressure less than 200 psig.

Report F-C4124 "Performance Qualification Tests of Four Valve Motor Operators" subjected the MO-7070 to 50 minutes of LOCA simulation at 240°F and 43.7 psig with tap water spray and relative humidity of 100%. Humidity was maintained at or near 100% by the following means: (1) use of saturated steam to obtain the initial temperature and pressure rise to 240°F, (2) use of fine mist water over the specimens, (3) use of saturated steam ejections to maintain chamber temperatures. MO-7070 failed the first test as a result of an electrical short across a

preexisting damaged terminal strip. The terminal strip was replaced with in-line insulated splices and successfully retested under the same environmental conditions. The like actuators in the plant were modified with the in-line splices. The motor of the tested Unit MO-7070 was replaced with a new unit in 1979 after the original failed due to unknown reasons.

The backup core spray valves, like the primary core spray valves (MO-7051 and -7061), operate within one hour maximum following a break. They may be required to close after recirculation has begun (maximum of 21 hours). In such a case, the primary spray line would provide core cooling.

Rotork has conducted testing for aging qualification. Technical Report TR 422 subjected an NA type actuator with a Class B insulated motor to an aging test which simulated 48 years at 60°C. The unit in the test was placed in a furnace for 100 hours at 180°C. Forty-eight years at 60°C was determined by the 10°C rule. TR 422 did not include a LOCA simulation and although an NA type actuator was used in the test, the only difference between the NA and 14A unit, according to is the motor housing material, cast iron versus cast aluminum. The results of the test, states, and the electrical characteristics will be exactly the same for the 14A as the NA type units. Test No. N/14/2 used a prototype unit built for material evaluation of the standard "A" and NA1 components. The actuator for MO-7070 and MO-7071 Type 14A are standard "A" units. The test results showed the "A" range components capable of withstanding 30 megarads during their 40-year life, according to The test unit, however, was equipped with a Class H insulated motor. actuators (Report B0003) with Class B motors have been irradiated to 20 megarads and have been subjected to outside containment HELB conditions for a 16-day period and have operated satisfactorily. Since the valves operate within one day and remain in position (open or closed) thereafter, the 30-day radiation dose of 0.[illegible]3 Mrads affects the aging of the components after it has completed its required function. The one-day dose of 0.49 Mrads is conservatively what the valve will be exposed to during its operating time. This dose is within the qualification dose of most materials.

In summary, based on the testing of the MO-7070 actuator and the tests of showing aging and radiation qualifications, the units are considered satisfactory for their intended service. Further, the operating time is less than one day, after which time the units will remain in position and not be required to actuate. Their motor starters located in the station power room, a nonharsh area, will not misoperate due to the environment they are in. The actuators are considered acceptable for use.

FRC EVALUATION:

The reference cited by the Licensee is summarized in the Response above. The test programs cited provide useful but not conclusive evidence that these MVAs will function adequately. However, the accelerated aging and nuclear irradiation exposures were conducted on a unit different than the one subjected to the long-term steam/spray exposure. Therefore, the results, while they do help to provide confidence that the performance under accident conditions will be satisfactory, are not conclusive and do not provide evidence of qualification. Because operation is required for approximately one day, FRC believes qualification should be established on a more rigorous basis. Also, the Licensee should make a conservative estimate of the qualified life, and develop a maintenance and equipment performance surveillance schedule to ensure that this equipment will perform properly under accident conditions.

FRC CONCLUSION:

This equipment belongs in NRC Category IV.b. Complete qualification has not been established, but the Licensee has provided sufficient evidence to show that the installed units are likely to function under accident conditions.

4.6 NRC Category V
EQUIPMENT WHICH 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 the following equipment items has been judged to be deficient or inadequate, based upon FRC's 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: (i) documentation reflecting qualification as specified in the DOR Guidelines has not been made available for review, (ii) documentation reflecting qualification has been made available for review and is totally inadequate, or (iii) the documentation indicates that the equipment item has not successfully passed required tests. The following equipment items are therefore considered unqualified.

4.6.1 Equipment Item No. 3
Electrical Penetration Inside Containment
Amphenol-Borg, Model Not Stated
(Original Licensee References 4-67 through 4-69;
Final Licensee References 36, 37, 1, and 79)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records, reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be

susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

Licensee References 4-67, 4-68, and 4-69 provide design information and radiation dose rate information but do not provide any evidence of environmental qualification. It is suggested that the Licensee contact the manufacturer to determine whether any test data is available demonstrating adequacy of the penetrations to meet the DOR Guidelines on environmental qualification.

LICENSEE RESPONSE:

Type 8: Coaxial Cable Penetration Assembly

The subject electrical penetrations are used for non-Class 1E circuits inside the containment. This qualification report, therefore, is submitted to prove the pressure boundary integrity of the penetration.

1. Coaxial Cable Penetration Assembly

This consists of an 8" pipe with a flange on the end to which a plate is bolted. The plate holds the coaxial cable connectors. The complete unit, as such, is welded to 8" IPS Schedule 40 nozzle, 12" long which, in turn, is welded to reactor building shell.

The connectors on the plant have 304 stainless steel shell which is press fit into and then welded to the plate.

These connectors have inserts made of glass material and are fused in placed by high heat process to achieve a pressurized seal. These inserts are suitable to handle three RG59 B/U cables. In addition, these connectors are provided with solder type mating connector with cable clamp and a protective cap on either side.

Therefore, the pressure boundary integrity of the complete penetration is based on the qualification of following components which are further discussed in subsequent paragraphs:

Glass Preform:

The glass preform which forms the primary pressure boundary for the penetration is made from Corning #9010 glass and is fused in place

by high heat process to achieve a pressurized seal. This glass begins to soften at 1160°F and is made up of 67% SiO_2, 7% A_2O, 7% K_2O, 5% Al_2O_3 and 12% BaO_2 plus 2% other. Document Reference 1 shows that this glass is similar to both silica glass and soda-lime glass. The table on Page 370 of Reference 36 indicates that silica glass is good to 2.5 x 10^8 rads. Further, based on the method of placement and the thickness of the glass, it is concluded that it can withstand 27 psig (41.7 psia).

Shell to Plate Assembly:

The connector shell is made of 304 stainless steel which is first press fit into the place and then welded. Such metal-to-metal assemblies should be able to withstand LOCA conditions of 235°F and 41.7 psia for a period of 30 days.

Gasket:

Upon analysis it has been determined that the gasket used between the mating surface of the fixed and the removable connectors is made of either the silicone rubber or the natural, butyl styrene butadiene rubber. Document Reference 1 on Pages 118-129 indicates that all rubbers can withstand at least 2 x 10^7 rads. Figure D-19 on Page 73 of Document Reference 37 shows that butadiene rubber (BUNA S) can withstand continuous temperature of 290°F.

2. The qualification information of the subject coaxial cable electrical penetration is based on the following documents and discussions (in Paragraph 2.1).

   a. Document Reference 1 on Pages 118 thru 129 and Page 370 indicates the radiation withstand cabilities of various rubbers and glass.

   b. Document Reference 37 on Page 73 (Figure D-19) confirms that butadiene rubber (BUNA S) can withstand continuous temperature of 290°F.

   c. Document Reference 79 on Page F-134 indicate that the melting point for glass used for sealing applications is not less than 1300°F.

3. Aging

Document Reference 36 and 37 provide evidence that silica glass or sodalime glass have much higher threshold for nuclear radiation (up to 2.5 x 10^8 rads) exposure than other dielectric materials and have very high temperature withstand capabilities (continuous operation of temperature up to 1000°F). Therefore, it is concluded that this fused glass is not susceptible to degradation due to

thermal or radiation aging and is considered to be qualified for 40-years' life at Big Rock Point Plant.

FRC EVALUATION:

Qualification testing as required by the DOR Guidelines has not been demonstrated. The Licensee has stated that the penetrations are qualified based on an analysis of the properties of the basic materials used in the penetrations; however, the Guidelines do not provide for the substitution of analyses for test. While such information is useful in selection of materials, the properties of the "as-manufactured" item need to be established by test.

The Licensee statements indicate that these penetrations are connected to non-safety-related equipment. Since such equipment may fail under LOCA or MSLB in containment, the combined effects of LOCA/MSLB and short circuit currents in the various conductors should be addressed.

FRC CONCLUSION:

The equipment belongs in NRC Category V because there is no evidence of qualification. From the information provided, FRC cannot determine whether the penetration integrity would be maintained in the event of LOCA.

4.6.2 Equipment Item No. 4
Electrical Penetration Located Inside Containment
Manufacturer and Model Not Stated
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

Component Electrical Penetration Type 9

This penetration, Type 9, was used in a research and development program for instrumental fuel assemblies. The penetration provided the electrical path from the fuel assembly to the computer.

This penetration was apparently somewhat different from other types. At the present time, the records for this penetration are being sought so

that a detailed analysis can be made for qualification. The qualification is directed toward containment boundary rather than electrical operability.

FRC EVALUATION:

The Licensee states that information on this penetration type is being sought from records and that qualification will be determined.

FRC suggests that if the research and development function is no longer required, the Licensee should evaluate capping the penetration rather than attempt to qualify it.

FRC CONCLUSION:

This item belongs in NRC Category V because there is no evidence of qualification.

4.6.3 Equipment Item No. 5
Flow Transmitters Located Inside Containment

Core Spray Flow (FT-2161 through 2164)
(Original Licensee References 4-2 and 4-13;
Final Licensee References 33, 38, and 61-63)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide environmental qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimen that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and

radiation aging. If the component contains such materials, a qualified life for the component must be established.

The Licensee's Reference 4-2 only lists the flow transmitters and states in the list that it is qualified to 41.7 psia and 235°F. There is no evidence of qualification by type testing or evaluation.

The Licensee pointed out that Equipment Item 14B, core spray flow transmitter, was subject to submergence conditions; however, there was no attempt to demonstrate operability under submerged conditions. Also, a manufacturer's certificate of compliance was cited as evidence of qualification; however, this is unacceptable according to the Guidelines.

FRC concludes that the Licensee should submit proper environmental qualification documentation.

LICENSEE RESPONSE:

The flow transmitters are Model 386. The flow transmitter is used to provide indication of core spray flow and also to detect if a core spray line is broken.

The environmental test used a Model Model differs from Model only in housing type. However, for the environmental test, the Model housing was used on the Model and, therefore, the test specimen is identical to that installed. The test sequence was to use steam to pressurize a test chamber to 60 psig. This pressure and a temperature of 288°F was maintained over test period and then cooled. The test instruments were read twice during the test and the readings matched that of the transmitters outside the chamber. The instruments were also separately tested for their ability to withstand radiation. They were exposed to 10^6 rads/h for 216 hours and passed. The total integrated dose far exceeds the qualification requirements. The transmitters are housed in a metal enclosure so that sprays will not affect them. Two hours after a LOCA, the temperature is at or below 160°F. The manufacturer states that the operational range of this transmitter is -40°F to 160°F. Therefore, the transmitter is able to withstand the effects of the LOCA because it was tested in a more severe environment and for the time period over which the containment conditions are more severe than the operational limits. No information has been found concerning age sensitivity. Should these transmitters fail, due to age degradation, failure will, in all probability, occur after the operator has ascertained the integrity of the core spray line. Once that is accomplished, core spray flow indication is not required for plant

shutdown. However, since core spray flow indication may be useful to the operator, age sensitivity will be investigated and appropriate action (partial replacement, full replacement, etc.) will be taken before June 30, 1982, assuming no procurement problems.

One of the flow transmitters will become submerged during the accident. The flow transmitter is considered qualified for submergence for the following reasons. Employees at the plant have stated that the transmitter and its leads are sealed. The transmitter was tested in a steam environment of 60 psig. This is far greater than the pressure the instrument will see even if submerged. It is believed that if steam at 60 psig cannot enter the transmitter to the extent of causing failure, then water at 27 psig will not result in failure.

FRC EVALUATION:

The Licensee has cited References 33, 38, 61, 62 and 63 as evidence of qualification. FRC has reviewed this documentation and has the following comments:

a. Reference 38 is a Product Bulletin. This reference is not evidence of qualification in accordance with the DOR Guidelines.

b. Licensee Reference 33 is a report of a steam exposure test of a pressure transmitter. With regard to this report, FRC comments:

   o The installed transmitter is a Model However, Reference 63 states that the Model was modified and later designated this model as Reference 62 states:

     "The first test utilized a standard industrial Model 332 transmitter. The test was unsuccessful. For the second test, they put the electronic package in an environmentally sealed case. This configuration was given a new model number, Model 386. The differential pressure unit, strain gage beam and electronic circuit board were all identical to those in the Model 332. The only difference is that the die cast aluminum case was replaced with an environmentally sealed case. Barton's numbering system required a new number for this and it was 386."

   However, FRC has further documented knowledge:

o

o

FRC concludes that this establishes the specific relationship between the test specimen (Barton Model 332-Mod I) and the Barton Model 386 high temperature pressure sealed electric transmitter; however, FRC must assume that Barton modification is equivalent to "Mod-I" as stated in FIRL C2667. Additionally, FRC notes that the power supply with the

c. FRC concludes that establishes satisfactory performance for the Barton transmitter with respect to qualification for radiation. Although the Licensee presented Reference 61 as evidence of qualification for radiation, FRC could not determine the applicability of this document.

d. As stated in the referenced test report, [33] the test chamber time-dependent temperature-pressure profile exceeded the postulated accident profile for but did not totally envelop the required environmental service conditions. The referenced test time duration, stated to be did not envelop the required accident profile 41-hour interval.

e. The Licensee states that one of the flow transmitters will become submerged, but since the transmitter was tested at 60 psig in a steam environment, this is far greater than the submerged pressure. However, the Guidelines require that specifying saturated steam as a service condition during type testing of equipment which becomes flooded in service is not an acceptable alternative for actually flooding the equipment during the test.

f. Pre-aging and qualified life have not been addressed.

g. The Licensee has not stated that the M297 power supply is located outside the adverse environment as per the test condition.

The Licensee has stated that age sensitivity will be investigated and that appropriate action will be taken before June 30, 1982.

FRC CONCLUSION:

This equipment item belongs in NRC Category V because evidence of qualification in the specific in-plant adverse environment has not been established beyond 2 hours for the required operating time (continuous) for the component. Also, submergence, aging, and qualified life have not been addressed. The Licensee has provided a justification for interim operation. FRC concludes that this justification appears acceptable. The Licensee stated that this component's age-sensitive materials will be investigated; however, a schedule for a definitive resolution has not been provided in accordance with NRC requirements.

4.6.4 Equipment Item No. 7
Level Switches Located Inside Containment

Monitors Steam Drum Low Water Level
(Original Licensee References 13-15, 4-18 and 66; Final Licensee References 92.4 and 18)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee has provided several references relating to the Yarway level indicating switch which monitors reactor water level. These references basically describe a test conducted in 1975 to demonstrate environmental qualification; however, this testing was done on a different model than that installed at the Big Rock plant. Also, a silicone sealant was used for a waterproofing compound for exposed electrical terminals. A hole was drilled in the bottom of the switch enclosure to prevent the enclosure's collapsing under external pressure; mercury switches were used and electrical connections for the lead entering the switch enclosure were waterproofed. The Licensee did not attempt to account for the differences in the model numbers or for the special features afforded the prototype test unit in relation to the actual units currently in operation at Big Rock Point. The DOR Guidelines state that the testing should be conducted on the same model number or an evaluation has to be performed to explain why the differences would not be significant.

FRC's review of the documentation leads to the conclusion that the level switches may be subjected to submergence conditions, and even though the Licensee has stated in Reference 4 that the switch's function of opening a motor-operated core spray system valve will only be needed for one minute of the postulated accident, there is no analysis or accident scenario performed

that would be able to provide reasonable assurance that the instrument would not be needed to provide a safety-related function for a longer term incident. Also, the design of the system's electrical circuit which initiates the core spray valve opening is one in which the level switch contact must open in order to perform its vital function. Since this switch has a high likelihood that water would create electrical tracking across the switch and thereby not allow the motor-operated valve to go to its open position, FRC concludes that having an unqualified switch performing this function is unacceptable.

Reference 18 provides the Licensee's analysis of type testing results which were conducted in 1975 and offers recommendations as to how the level switch can be altered in order to provide a more qualified instrument to survive the containment's service conditions. Their recommendations included the addition of silicon sealant on terminal block connections, the replacement of the indicator dial, diffuser, and case with Lexan materials (since the indicator had become distorted during the type test and prevented the operation of one of the level switch contacts), the replacement of the plastic plugs with metal plugs, and the changing of the mercury switches with an Acro switch in the instruments. The Licensee has not provided any positive indication of what modifications may have been made or if any followup testing was conducted after any of these revisions to the instrument. FRC concludes that the Licensee should provide evidence as to what type of modifications have been performed and how they have resulted in assuring the unit is environmentally qualified.

The Licensee did not address the DOR Guidelines requirement that materials subject to thermal or radiation degradation be tested or evaluated. In addition, there were no statements regarding qualified life of the instrument. A total material listing was not provided which may have otherwise been able to remove concerns as to the instrument's use of elastomers or phenolics that previous testing has shown to be susceptible to thermal stresses.

FRC concludes that the Licensee should provide additional qualification documentation to establish long-term environmental credibility since several materials in the level switch, such as the terminal blocks and non-metal components, may have significant degradation occur at elevated temperatures.

The tests which were conducted on the were limited because they did not address aging, submergence, external pressure, chemical spray, or radiation. The temperature-humidity profile which was employed in the test was not of sufficient duration to envelop the Big Rock Point plant containment profile condition.

In summary, FRC concludes the following:

a. 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 to those of the installed equipment. The similarity reviewed by FRC thus far does raise additional concerns regarding operability of this manufacturer's level switches.

b. The test specimen was not subjected to thermal aging. The Guidelines state that thermal aging of test specimens is required if the component contains materials known to be susceptible to significant degradation due to aging. The materials used in this equipment have not been analyzed for susceptibility to aging degradation. Therefore, the test result cannot be considered conclusive.

c. The test specimen was not exposed to nuclear radiation to simulate DBE conditions, nor was information submitted to demonstrate that the materials used would not be degraded by exposure to nuclear radiation, as is required by the Guidelines.

d. The level switch has the possibility of being subjected to submergence conditions; however, no testing was performed nor was a suggestion made by the Licensee to enclose the units in waterproof enclosures.

e. Modifications which may have been made to the unit have not been positively identified and any qualification upgrading cannot therefore be credited.

The documentation submitted to date must therefore be regarded as inconclusive and the Licensee should provide additional documents to establish qualification.

LICENSEE RESPONSE:

Component Level Switch LS-RE06A,B and LS-RE20A,B

Level switches LS-RE06A,B and LS-RE20A,B provide the necessary signal to the reactor protection system for a reactor scram due to low water level in the steam drum. In addition to the function described above, the LS-RE20A,B units are fitted with a differential transformer and armature rod assembly driven by the primary instrument pointer to provide that an electrical signal can be sent to slave indicators located in the control room.

The required operating time for these instruments has been established at 10 minutes. The basis for this time is that there are some small sized

steam line breaks that may occur which will not provide an automatic scram due to low steam drum level or containment high pressure. This is because (1) the water inventory lost through the break is maintained by the feedwater system and (2) the containment is normally vented to the atmosphere via the stack such that for small breaks the containment high pressure set point may not be reached. These break sizes involve leakages of 75#/s or less. No credit for operator action is taken for the first 10 minutes of a steam line break event. After 10 minutes it is reasonable to expect the operator to take the action necessary to manually shut down the reactor. For break sizes resulting in flow rates greater than 75#/s, the level switches must remain operable to automatically scram the reactor. A discussion of the small break scenario was provided to NRC on August 26, 1980 as noted in Reference 66. It is worthwhile to note that from the data submitted as Reference 13, the most limiting steam line break size results in a flow rate of 75#/s. With this flow rate, containment temperature was calculated to reach 217°F after 10 minutes rather than the peak LOCA temperature of 235°F.

If the reactor scrams for high containment pressure via PS-664 through PS-667, then there is no need for LS-RE06A,B and LS-RE20A,B since they are only needed for the scram function. Thus, the pressure that the level switches need be qualified for is that corresponding to the set point of the pressure switches PS-664 through PS-667, namely, 16.4 psia.

The maximum radiation dose that level switches must reasonably endure is 4.9×10^3 rads. The basis for this dose valve is that if the full 10-minute time interval is needed, the drum level is being maintained by normal feedwater addition. In this case, there would be no core uncovery with resulting fuel damage. The radiation dose assigned for a steam line break in the pipe tummel assumed a total loss of the primary coolant inventory with the highest level of coolant activity allowed by Technical Specifications. The small steam line break event is similar to the MSLB outside containment with respect to the source of radiation and therefore the pipe tunnel dose was used. For the larger break sizes, the required operating times are very much shorter, thus less dosage is incurred by the level switches.

The level switch vendor, Yarway, was asked to provide a listing of organic materials used in the Model 4320PE level switches. The vendor responded with the following for the 4300 series:

1. Diaphragm - Neoprene Coated Dacron Fabric

2. Backing Plate Gasket - Klingerit (Compressed Asbestos)

3. Dust Plug Grommet - Neoprene

4. Light Diffuser - Molded Translucent Plexiglas

5. Front Cover - Molded Clear Plexiglas

6. Dial - Translucent White Vinyl

NOTE: Items 4, 5 and 6 above have been replaced with Lexan at Big Rock Point.

7. Casing Gasket - Cork and Rubber Compound DK-149

8. Switch Spacers - Laminated Melamine

9. Switch - Mercury

10. Relays - BO Type - Allied Control Co.

Other organic materials found in the armature rod assembly and differential transformer and guide assembly include:

11. Armature Bushing - Synthane Grade L

12. Differential Transformer and Guide Assembly - Copper Wire With Silver Plated Teflon Insulation (MIL-W-16878, Type E), Epoxy CPI-4270 Hysol, Markol Corp. Shrink Tubing HT105C and HF105

13. Terminal Blocks - Molded Phenolic Base

14. Coil Bobbin - Nylon

15. Winding - Nyclad Wire

16. Disc - Synthane Grade LE

17. Sealer - Epoxy

18. Insulator - Fish Paper 0.010" Thick

The level instruments need to be qualified for a rise in temperature from ambient to 217°F over a period of 10 minutes. The Model 4300 series are similar in design and materials of construction as the Model 4400 series. Various tests (References 2, 3 and 4) were performed on different 400 series instruments covering a range of peak temperatures from 212°F to 269°F, with a peak pressure of 44.7 psia and 100% relative humidity. These tests provide the necessary assurance that LS-RE06A,B and LS-RE20A,B will remain operable until their intended function is complete.

The spray water in containment is taken from Lake Michigan which is relatively pure. The level instruments are located on a concrete wall below the containment spray header; however, the nozzles are oriented in the horizontal direction so that the instruments will not be subjected to

direct impingement of the spray. Further, the instruments are mounted in a metal cabinet with a top that will protect the instrument from spray.

The most susceptible material to the radiation dose the instruments must endure is the silver plated Teflon wire. According to the Nuclear Engineering Handbook by H. Etherington, et al., Page 10-144, Teflon-insulated wire suffers a 25% change in elongation at a radiation dose of 3×10^4 rads. Thus, these level switches are deemed qualified for radiation.

Although the ambient temperatures are on the average low (70°F), it is recognized that some of the components in the instruments may be susceptible to thermal aging. The LOCA event will contribute little to the overall aging because of the short time the equipment is required to function. Further investigation will be performed to determine which materials are age sensitive and whether or not failure of these parts will affect instrument operation.

To summarize, it is concluded LS-RE06A,B and LS-RE20A,B will satisfactorily perform their intended function during the LOCA event.

FRC EVALUATION:

The Licensee has provided an equipment analysis as evidence that the level switch is qualified for pressure temperature, and radiation environmental service conditions for a time period of 10 minutes following a postulated LOCA or MSLB occurrence in the containment. The 10-minute evaluation is not satisfactory, however, because the NRC requires that safety-related electrical equipment be shown to be operable for at least 1 hour plus the time for which it must function in that equipment's environmental service condition. The one-hour operational qualification requirement has not been demonstrated by the Licensee in its analysis approach.

In addition, the Licensee did not provide evidence to address the following concerns identified in the DITER:

a. Electrical circuit analysis to review fault in electrical contact or terminal block

b. Replacement of plastic plugs and mercury switch as recommended by Reference 4-18

c. Follow-up testing

d. Model number correlation between tested units and the level switch located in the plant. (A simple statement stating they are similar is insufficient for FRC verification. Specific differences need to be identified and discussed.)

e. Silicone sealant on terminal board connections.

The radiation component analysis provided by the Licensee did not identify all of the constituent's radiation threshold values for a complete comparison with the radiation environmental service condition of approximately 0.1 Mrad for a 1-hour post-accident gamma dosage. The most susceptible component according to the Licensee is silver-plated Teflon wire which has a theshold value of 0.03 Mrad and clearly falls below the environmental exposure of 1 hour.

FRC CONCLUSION:

This level switch belongs in NRC Category V because qualification documentation has not been provided to demonstrate operability for at least 1 hour and qualified life has not been addressed.

4.6.5 Equipment Item Nos. 38A, 38B, 39A, 39B and 40
Terminal Blocks Located Inside Containment
38A: General Electric Model CR-151
38B: States Co. Model NT
39A: Westinghouse Electric Model 542247
39B: Westinghouse Electric Model 805432
40: General Electric Model EB-25
(Original Licensee References 4-14 and 4-32; Final Licensee References 36, 75, 25, 42, 40, 76, 26)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class IE equipment. Type testing is the preferred method of qualification for Class IE electrical equipment

which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

References 4-14 and 4-33 do not provide any test information on this equipment.

The Licensee should determine from the supplier if any qualification documation is available.

LICENSEE RESPONSE:

Component Terminal Blocks (Item No. 38A and 38B)

The GE Type CR-151 and States Company Type NT terminal blocks were tested as part of GE's electrical penetration qualification test program.

The terminal blocks were mounted in a test chamber. The insulation resistance was measured using 500 V dc at ambient conditions. The blocks were then subjected to a LOCA environment during which the insulation resistance was measured once per day. The test sequence was as follows: The environment was held at 260°F and 21 psig for 1.5 days (pre-aging); then the pressure and temperature were increased to 320°F and 75 psig and held for 1.5 hours; increased to 340°F and 130 psig and held for 3 hours; decreased to 320°F and 75 psig and held for 4.5 hours and then decreased to 260°F and 21 psig and held for 8 days. The pre-aging simulates 40-year life for most materials used in terminal blocks. The temperature and pressure greatly exceed that needed to demonstrate qualification and the test duration is longer than required. Reference 1 also stated that the insulation resistance was lower but remained at a sufficient level to assure continued function of the electrical equipment without circuit overload.

Spray was not used during the test. These terminal blocks are located in sealed junction boxes and, therefore, are not affected by the sprays (see junction box write-ups). The radiation at the center of containment is 7.3×10^5 rads over 30 days. The terminal blocks are in sealed

junction boxes. The Big Rock Point containment is split into many rooms separated by concrete walls. The terminal blocks are mounted on these walls. Also, ambient conditions are reached in 3 days, not 30. For these reasons, the total integrated dose these blocks will see is much less than 7.3×10^5. The terminal blocks will be able to withstand these relatively low doses as most of the materials of construction can withstand 10^6 rads.

The terminal blocks were tested in an environment more severe and of longer duration than that required to demonstrate qualification. The blocks were preaged prior to the test and are evaluated to be able to withstand the radiation. Therefore, they are qualified.

Component Terminal Blocks (Item No. 39A and 39B)

The terminal blocks are cellulose filled phenolic made by Westinghouse. Model 542247 is the same as Model 805432 but with a black marking strip per Reference 42. At Big Rock Point, all terminal blocks are covered by splashproof metal covers.

These blocks were environmentally tested. The blocks were mounted to a fixture which was attached to a header through which copper feed throughs were passed. The test consisted of applying 600 V ac to the terminals and filling the test chamber with steam. Borated water was sprayed at the rate of 0.32 gpm for 1 hour. The environmental conditions were as follows: 340°F and 106 psig for 2 hours; decreasing to 329°F and 93 psig at 3 hours (borated water introduced at this time); rising slightly to 332°F and 91 psig at 4 hours; maintained until 5.5 hours; allowed to cool (at 21.5 hours the temperature was still above ambient) and opened at 26.5 hours. The insulation resistance decreased but the blocks were able to function at 600 V ac. The test duration does not cover the time until ambient is reached at Big Rock Point but is still considered to be an applicable test because the temperatures reached were much higher than that at Big Rock Point and for a much longer period of time. Also, the test report pointed out that at 21.5 hours, the temperature was still above ambient. At this time, during the Big Rock Point LOCA, the temperature pressure envelope shows the temperature at 130°F; however, the actual analytical break curves would show temperature below that. Neither of these temperatures are significantly above ambient so that the test very closely matched the end of the temperature profile. Lastly, the materials of construction are not age sensitive. Reference 40 shows that the qualified life is 40 years at 230°F. Therefore, a shorter test duration is acceptable.

The materials of construction were evaluated in Reference 40 to be capable of withstanding 2×10^7 rads. This is much higher than the gamma dose received in an accident. Beta radiation will not affect the terminal blocks as they are housed in splashproof housings. The fact that the spray lasted 1 hour shows that these blocks can withstand the spray. The blocks at Big Rock Point are housed in junction boxes and, therefore, not adversely affected by sprays.

The terminal blocks passed an environmental test with conditions exceeding those found at Big Rock Point for a duration which was judged acceptable. The terminal blocks contain no age sensitive materials and can withstand the radiation. Therefore, these blocks are qualified.

Component Terminal Blocks - General Electric, Westinghouse (Item No. 40)

The terminal blocks were tested by Northeast Utilities. The blocks tested were a GE Type EB-25 made of wood-flour-filled phenolic and a Westinghouse Style 805432 made of the same material. These blocks were mounted both horizontally and vertically in covered boxes. The terminal blocks at Big Rock Point are also in covered boxes.

The test sequence was as follows: First, the blocks were thermally aged at 150°C for 171 hours. According to the Arrhenius plot, this is equivalent to 40 years at 158°F. Since ambient condition at Big Rock Point is much less than this, the blocks will withstand the aging effects of 40 years plus LOCA. The terminal blocks were then irradiated to 5×10^6 rads. This exceeds the radiation level required to demonstrate qualification.

The environmental test was conducted by using steam to pressurize and heat up the test chamber. The chamber conditions after 8 seconds were 286°F and 40 psig which exceed the accident conditions. These conditions were held for 15 minutes then, to simulate recirculation, the pressure was lowered to 35 psig, held for 60 seconds, then raised to 286°F and 40 psig and held for 31 hours. The conditions were lowered to 232°F and 7 psig over 3 hours and held for 101 hours. These conditions envelop the accident conditions. The blocks passed the test.

Since the blocks were tested to environmental conditions more severe than that at Big Rock Point and were thermally aged prior to the test, it is considered that these are qualified.

FRC EVALUATION:

a. For Item No. 38A and 38B, the Licensee states that the terminal blocks are in sealed junction boxes and refers to the writeup on junction boxes which states that to equalize differential pressure all boxes were provided with vents to the containment atmosphere. FRC cannot determine from the information provided whether the junction boxes for Item Nos. 38A and 38B are sealed pressure-proof units or vented to the containment atmosphere.

b. The test reports information provided by the Licensee has been reviewed, as well as the analyses presented in the note for Items 38A, 38B 39A, 39B, and 40. FRC has the following comments:

1. The testing discussed in the references does not conform to the Guidelines requirements and it therefore has not been shown by test that the combined effects of thermal aging, radiation, and steam/chemical spray environments postulated to follow a LOCA event are unlikely to cause terminal block failure.

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) consider the effects of beta radiation and the proximity of the installed equipment to the sump or other concentrated sources of radiation.

3. Aging degradation has been addressed. However, 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-reinforced 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 Mrad and that elongation and impact strength are reduced by 25% at 3 to 8 Mrads.

   The mechanical and thermal properties of wood-flour-filled phenolics are highly variable as shown in Appendix D. 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.

4. During one of the steam exposure tests in the program described in the tests, for NUSCO, a short circuit developed on one of the blocks being tested because the screw that attaches the block to the junction box had been tightened to the point where the rather brittle cellulose-filled phenolic had cracked. This suggests that the overall qualification is quite sensitive to the mounting procedure and technique used. No documentation has been provided showing that this potential concern was addressed during the installation of this equipment.

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 a 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 6mm in diameter, but in some other reactors are 25mm 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 statistical evaluation of results for probability of failure as a function of time and voltage.

FRC CONCLUSION:

These equipment items belong in NRC Category V because there is no assurance that the terminal blocks would perform reliably or transmit reliable instrument signals under LOCA or HELB conditions. If the junction boxes for Item 38 are sealed and satisfactory for pressure conditions under LOCA, these items would be considered to be NRC Category II.a.

4.6.6 Equipment Item No. 9
Level Transmitter Located Inside Containment
ITT-Barton Model 386
Containment Water Level (LT-3171)
(Licensee References 33, 38, and 61-63)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

The level transmitter is a Model 386. The transmitter is used to provide indication of containment water level to the operator prior to entering the recirculation mode. This transmitter works in parallel with LS-3562 through LS-3565. Model 332 was environmentally tested. The environmental test used a Model 332. Model 332 differs from Model 386 only in housing type. However, for the environmental test, the Model 386 housing was used on the tested Model 332 and, therefore, the test specimen is identical to that installed. The test sequence was to use steam to pressurize a test chamber to 60 psig. This pressure and a temperature of 288°F was maintained over a two-hour test period and then cooled. The test instruments were read twice during the test and the readings matched that of the transmitter outside the chamber. The instruments were also separately tested for their ability to withstand radiation. They were exposed to 10^6 rads/h for 216 hours and passed. The total integrated dose far exceeds the qualification requirements. The transmitters are housed in a metal enclosure so that sprays will not affect them. Five and one-half hours after a LOCA, the temperature is at or below 160°F. The manufacturer states that the operational range of this transmitter is -40°F to 160°F. Therefore, the transmitter is able to withstand the effects of the LOCA because it was tested in a more severe environment and for the time period over which the containment conditions are more severe then the operational limits. No information has been found concerning age sensitivity. Should these transmitters fail due to degradation, containment level indication can be obtained by use of LS-3562 through LS-3565. As part of the modifications, as a result of TMI, a new fully qualified level transmitter will be installed.

FRC EVALUATION:

The Licensee has cited References 33, 38, 61, 62, and 63 as evidence of qualification. FRC has reviewed this documentation and has the following comments:

a. Reference 38 is an ITT Barton Product Bulletin. This reference is not evidence of qualification in accordance with the DOR Guidelines.

b. Licensee Reference 33 is a report of a test of a pressure transmitter. With regard to this report, FRC comments:

- The installed transmitter is a Barton Model 386. However, Reference 63 states that the Model 332 was modified and later Barton designated this model as 386. Reference 62 states:

  "The first test utilized a standard industrial Model 332 transmitter. The test was unsuccessful. For the second test they put the electronic package in an environmentally sealed case. This configuration was given a new model number, Model 386. The differential pressure unit, strain gage beam and electronic circuit board were all identical to those in the Model 332. The only difference is that the die cast aluminum case was replaced with an environmentally sealed case. Barton's numbering system required a new number for this and it was 386."

  However, FRC has further documented knowledge

-

-

FRC concludes that this establishes the specific relationship between the test specimen (Barton Model 332-Mod I) and the Barton Model 386 high-temperature pressure-sealed electric transmitter; however, FRC must assume that Barton modification is equivalent to "Mod-I" as stated in FIRL C2667. Additionally, FRC notes that the power supply with the

c. FRC concludes that WCAP-7410-L establishes satisfactory performance for the Barton transmitter with respect to qualification for radiation. Although the Licensee presented Reference 61 as evidence of qualification for radiation, FRC could not determine the applicability of this document.

d. As stated in the referenced test report, [33] the test chamber time-dependent temperature-pressure profile exceeded the postulated accident profile for but did not totally envelop the required environmental service conditions. The referenced test time duration, stated to be did not envelop the required accident profile 41-hour interval.

e. Pre-aging and qualified life have not been addressed.

f. The Licensee has not stated that the M297 power supply is located outside the adverse environment as per the test condition.

The Licensee has stated that age sensitivity will be investigated and that appropriate action will be taken before June 30, 1982.

FRC CONCLUSION:

This equipment item belongs in NRC Category V because evidence of qualification in the specific in-plant adverse environment has not been established beyond 2 hours for the required operating time (continuous) for the component. Also, aging and qualified life have not been addressed. The Licensee has provided a justification for interim operation. FRC concludes that this justification appears acceptable. The Licensee stated that this component's age-sensitive materials will be investigated; however, a schedule for a definitive resolution has not been provided in accordance with NRC requirements.

4.6.7 Equipment Item No. 48
Electrical Cables Located Inside Containment
Manufacturer and Model Not Stated
(Licensee References 53, 57, and 58)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE REPSONSE:

Component Cable Types 21 and 24

Type 21 - 1/C #12 AWG 7 Strand 600 V Cable With Butyl Insulation and PVC Jacket

Type 24 - 1/C #6 AWG 7 Strand 600 V Cable With Butyl Insulation and PVC Jacket

Type 22 - 3 - 1/C #12 AWG 7 Strand 600 V Cable With Butyl Insulation and PVC Jacket

The subject cables are used in Class 1E circuits inside the containment. The qualification is based on the following documents and discussion:

A. Per Document Reference 53, the butyl insulation can withstand a radiation dose of up to 5×10^6 R with no change in dielectric property and 5×10^7 R with 20% reduction in dielectric property. The 30-day integrated radiation dose during accident is 7.3×10^5 R (gamma) and 1.3×10^7 (beta). The beta radiation decreases by an order of 10 for approximately 30 mil thickness or insulation. Thus, allowing for cable jacket, conduit and tray cover, etc, the total radiation dose at the cable insulation will be less than 5×10^6 R. This meets the environmental requirements on radiation.

B. Per Document Reference 58, cable with butyl insulation can withstand 125°C (275°F) for four weeks with no change in dielectric property. Also, the dielectric strength relatively remains constant after exposing to air-oven aging test at 121°C (250°F) for 12 months.

C. Per Document Reference 58, the physical property of cable is within acceptable limits after exposing to air-oven test at 121°C for 28 months continuously and for 60 months with alternate seven days at room temperature. This accelerated aging test parameters far exceed the 40-year life requirement for cable in an environment where the normal ambient is 80°F. This approach is based on ten degree half-life rule per IEEE-101 (1974). Document Reference 57 also supplements this data where the test was done at 121°C for 293 weeks.

E. Per Document Reference 57, the cable passed the dielectric test after immersion in water for 160 weeks and, therefore, will be adequate in the 100% humidity and containmennt spray environment. Big Rock Point containment sprays are lake water and noncorrosive.

F. No credit has been taken for the jacket furnished.

G. There is no difference in pressure between any two terminations of a particular cable inside the containment and the effect of pressure on the ability of the cable to perform during an accident will not be impaired.

FRC EVALUATION:

The Guidelines do not permit an analysis in lieu of testing equipment. FRC is not aware of any testing of the cables represented by this equipment item. The Licensee submittal has not provided conclusive evidence that the safety function will be performed.

The Guidelines require identity between tested and installed equipment.

a. The Licensee assertion that because there is some data on insulating basic raw material the installed cables are qualified is not supported by the Guidelines, IEEE-323, or IEEE-383 and is not correct.

b. FRC is not aware of any testing which demonstrates qualification to the Guidelines requirements.

FRC CONCLUSION:

This equipment belongs in NRC Category V because there is no evidence of qualification. The installed cable should either be tested to demonstrate compliance with the Guidelines or replaced.

4.6.8 Equipment Item No. 49
Electrical Cables Located Inside Containment
Manufacturer and Model Not Stated
(Licensee Reference 54)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE REPSONSE:

600 Volt Cables With Sizes as Detailed Below With PVC (Polyvinyl Chloride) Insulation and Jacket

Type 29 - 3/C #12 AWG 7 Strand

Type 30 - 1/C #12 AWG 7 Strand

The above cables are used in Class 1E circuits and are qualified for use inside the containment as below:

A. Per Table 1 and Page 3 of Document Reference 54, the PVC insulation exhibits satisfactory physical properties (tensile strength and elongation) when exposed to gamma radiation up to 5×10^7 R. At 5×10^6 R, the tensile strength increases to 104% of its initial value (before radiation) and percent retention for elongation increases to 115%. When taking credit for the jacket, conduit and tray cover to absorb the beta radiation, this withstand value meets or exceeds the requirements.

B. From Page 4 and Table VI of Document Reference 54, PVC insulating material can retain (tested when immersed in water) a dielectric strength in excess of 86% of its initial value when subjected to 5 x 10^6 R and 72% when subjected to 5 x 10^7 R.

The subjected 600 V cables are used in 125 V dc circuits and there is conservative margin in dielectric strength for continued operation even after exposure to a radiation dose of 5 x 10^7 R.

C. From Page 3 and Table II of Document Reference 54, the PVC can withstand a temperature of 200 hours at 136°C (245°F) and has a useful life in excess of 40 years. This tested value is in excess of the parameters required by ICEA (formerly IPCEA) standards for air oven test; ie, 121°C for 168 hours.

D. The peak temperature inside the containment of 235°F after an accident at Big Rock exists for less than one hour and decreases to 80°F in less than three days. The withstand temperature and duration exceeds the requirements.

E. There is no difference in pressure between any two terminations of a particular cable inside the containment and the effect of pressure on the ability of the cable to perform during an accident will not be impaired.

F. Per procurement documents, the subject cables were manufactured to meet ICEA (formerly IPCEA) standards including the accelerated water absorption test requirements.

1. Paragraphs A, B and F provide evidence for radiation, humidity, water spray and submergence qualifications.

2. Paragraphs C, D and E provide evidence for temperature, pressure and aging qualifications.

FRC EVALUATION:

The Guidelines do not permit an analysis in lieu of testing equipment. FRC is not aware of any testing of the cables represented by this equipment item. The Licensee submittal has not provided conclusive evidence that the safety function will be performed.

The Guidelines require identity between tested and installed equipment.

a. The Licensee assertion that because there is some data on insulating basic raw material the installed cables are qualified is not supported by the Guidelines, IEEE-323, or IEEE-383 and is not correct.

b. FRC is not aware of any testing which demonstrates qualification to the Guidelines requirements.

FRC CONCLUSION:

This equipment belongs in NRC Category V because there is no evidence of qualification. The installed cable should either be tested to demonstrate compliance with the Guidelines or replaced.

4.6.9 Equipment Item No. 51
Electrical Cables Located in Electrical Penetration Room
Manufacturer and Model Not Stated
(Original Licensee References 55, 57, 58)

ORIGINAL TEXT TAKEN FROM INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

Type 21 - 1/C 312 AWG 7 Strand 600 V Cable with Butyl Insulation and PVC Jacket
Type 22 - 3 - 1/C #12 AWG 7 Strand 600 V Cable with Butyl Insulation and PVC Jacket
Type 24 - 1/C #6 AWG 7 Strand 600 V Cable With Butyl Insulation and PVC Jacket

The subject cables are used in Class 1E circuits outside the containment. The qualification is based on the following documents and discussion:

A. Per page 534 in Document Reference 55, the butyl insulation can withstand radiation up to 5×10^6 rads with no change in dielectric property. This exceeds the environmental requirements on radiation.

B. Per Document Reference 58, cable with butyl insulation can withstand 125°C (257°F) for four weeks with no change in dielectric property. Also, the dielectric strength relatively remains constant after exposing to air-oven aging test at 121°C (250°F) for 12 months.

C. Per Document Reference 57, the physical property of cable is within acceptable limits after exposing to air-oven test at 121°C for 28 months continuously and for 60 months with alternate seven days at room temperature. These accelerated aging test parameters far exceed the 40 year-life requirement for cable in an environment where the normal ambient ranges from 5°C to 38°C (40°F to 100°F). This

approach is based on ten degree half-life rule per IEEE-101 (1974). document Reference 3 also supplements this data where the test was done at 121°C for 293 weeks.

D. Per Document Reference 3, the cable passed the dielectric test after immersion in water for 160 weeks.

E. No credit has been taken for the jacket furnished.

FRC Evaluation:

The Guidelines do not permit an analysis in lieu of testing equipment. FRC is not aware of any testing of the cables represented by this equipment item. The Licensee submittal has not provided conclusive evidence that the safety function will be performed.

The Guidelines require identity between tested and installed equipment.

a. The Licensee assertion that because there is some data on insulating basic raw material the installed cables are qualified is not supported by the Guidelines, IEEE-323, or IEEE-383 and is not correct.

b. FRC is not aware of any testing which demonstrates qualification to the Guidelines requirements.

FRC CONCLUSION:

This equipment belongs in NRC Category V because there is no evidence of qualification. The installed cable should either be tested to demonstrate compliance with the Guidelines or replaced.

4.6.10 Equipment Item No. 50
Electrical Cables Located Inside Containment
Manufacturer and Model Not Stated
(Licensee References 56, 59, and 83)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE REPSONSE:

1. Component: 600 volt power and control cable with polyethylene insulation with polyvinyl chloride jacket. Cables are of the following types:

Type 2 - 3/C #14 AWG

Type 3 - 5/C #14 AWG

Type 4 - 7/C #14 AWG

Type 5 - 2/C #14 AWG

2. The above cables are used in Class 1E circuits inside the containment. The cable will be exposed to lake water spray and submergence.

2.1 The qualification information of the subject cable is based on the following documents and discussions:

2.1.1 Temperature

Document Reference 83, Chapter D, Table D-22, Page 71, indicates that the highest usable continuous temperature for polyethylene is 212°F.

Ambient temperature at Big Rock containment is only 90°F.

Engineering judgment shows that cable will be able to withstand 235°F for maximum period of one hour during LOCA.

This is based on the reasoning that the continuous withstand temperature is 212°F and a marginal increase in temperature to 235°F for one hour will only decrease its thermal life and not degrade its performance.

2.1.2 Humidity and Spray

Document Reference 56 on Page 535, Table IX, states that HD polyethylene does not show any signs of degradation when subjected to 90°C water for durations exceeding nine weeks. Based on this, it is concluded that the cable will be able to function satisfactorily under 100% humidity and water spray conditions. Big Rock Point containment spray water is noncorrosive lake water.

2.1.3 Radiation

a. Document Reference 59, Section 3.3, Table 3.3, Page 121, indicates that there will be no "measurable effects" on polyethylene on exposure to radiation of 10^7 rads.

b. Document Reference 83 on Page 535, Table IX, states that irradiated cable (5×10^7 R) can withstand immersion in 90°C water for more than nine weeks without electrical failure so

radiation dose of 5×10^7 rads exceeds the total requirement of 1.373×10^7 rads (beta 1.3×10^7 R and gamma 7.3×10^5 R).

c. Cable jacket, trays and conduits provide shielding against beta radiation. Therefore, the conductor insulation will be exposed to a significantly lower dose of beta radiation.

2.1.4 Pressure

Review of the post-LOCA test reports for various types of cables indicates that pressure-related cable failures are extremely rare. Therefore, even though there is no test data available to prove pressure withstand, it appears very unlikely that cable failure can occur due to short duration peak pressure of the order of 41.7 psia.

2.1.5 Test Specimen Aging

a. Per Document Reference 83, Page 535, Table IX, irradiated (5×10^7 rads) cable can withstand immersion at 90°C water for more than nine weeks.

b. Document Reference 83, Chapter D, Table D-22, Page 71, indicates that the highest usable continuous temperature for polyethylene is 212°F.

Based on the above references, it would appear that the cable is not generally susceptible to significant radiation induced or heat induced aging degradation.

2.1.6 Submergence

Document Reference 83, Page 535, Table IX, indicates that irradiated (5×10^7 rads) cable successfully passed dielectric tests after immersion in 90°C water for nine weeks.

FRC EVALUATION:

The Guidelines do not permit an analysis in lieu of testing equipment. FRC is not aware of any testing the cables represented by this equipment item. The Licensee submittal has not provided conclusive evidence that the safety function will be performed.

The Guidelines require identity between tested and installed equipment.

a. The Licensee assertion that because there is some data on insulating basic raw material the installed cables are qualified is not supported by the Guidelines, IEEE-323, or IEEE-383 and is not correct.

b. FRC is not aware of any testing which demonstrates qualification to the Guidelines requirements.

FRC CONCLUSION:

This equipment belongs in NRC Category V because there is no evidence of qualification. The installed cable should either be tested to demonstrate compliance with the Guidelines or replaced.

4.6.11 Equipment Item No. 52
Electrical Cables Located Inside Containment
General Electric, Model Not Stated
(Original Licensee References 9, 56, and 83)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE REPSONSE:

1.0 Component: (1) #14 AWG control cable GE Type RHW 75°C Versotol (styrene butadiene rubber) insulation and geoprene jacket. Versotol is also known as GR-S or BUNA S and (2) #8 AWG power cable GE Type RHW 75°C Versotol insulation and geoprene jacket.

2.0 The above cable is used in Class 1E penetration from inboard to outboard. The cable will not be exposed to containment spray or submergence.

2.1 The qualification information of the subject cable is based on the following documents and discussion:

2.1.1 Temperature

a. Document Reference 83, Figure D-19, which shows the high-temperature limits of the various rubbers, indicates that BUNA S can withstand up to 290°F.

b. Document Reference 56 indicates that the irradiated (5×10^7 R) cable can withstand a combined high-temperature and steam environment for four days at 40 psig and 142°C (287.6°F).

These values of 290°F and 287.6°F are higher than the 235°F (peak) for one hour required for the Big Rock Point Plant.

2.1.2 Humidity

Document Reference 56, Page 534, describes the LOCA test on irradiated cable specimens. The specimens were exposed to a radiation dose of 5×10^7 R and then placed in water-filled jars. The jars were inserted into a steam autoclave and subjected to a temperature of 142°C (288°F) and 40 psig (54 psia) for four days. Results of the test, tabulated on Table IX, Page 535, of Reference 1 indicate that the specimen with SBR insulation successfully passed dielectric test at 80 V/mil at 90°C (194°F). The BRP LOCA profile calls for a peak temperature of 235°F, 100% humidity and peak pressure of 41.7 psia for one hour, dropping to 140°F and 25 psia in 25 hours. After three days, temperature drops to the normal ambient of 80°F.

Because the test temperature, humidity, pressure and duration exceeded the BRP LOCA requirements, it is concluded that the cable is qualified to withstand the BRP LOCA environment.

2.1.3 Radiation

Document Reference 56 in Table XI indicates the threshold of gamma radiation damage (rad) for elastomer-based cable insulations. The table states that cable with SBR insulation can withstand doses up to 5×10^7 rad gamma.

It is concluded that the ca[illegible]is qualified to Big Rock requirements based on the f[illegible]ing evaluation:

a. Reference 56 indicates that the cable can withstand 5×10^7 rad gamma which exceeds the requirement of 7.3×10^5 R gamma and also exceeds the total requirement of 1.373×10^7 rads (1.3×10^7 R beta and 7.3×10^5 R gamma).

b. Cable jacket provides shielding against beta radiation. Therefore, the conductor insulation will be exposed to significantly lower dose of radiation.

2.1.4 Pressure

It can be concluded that the cable is qualified to Big Rock Point requirement based on the following:

a. Document Reference 9 includes a report on final integrity test and leakage rate determination for reactor containment of the Big Rock Point Plant. This report indicates a satisfactory leakage rate of 0.21% per 24 hours at 10 psig and further concludes that the leakage at design pressure of 27 psig would be in the same order as the leakage at 10 psig. The report states "it is safe to say that the leakage at design pressure

is far below the value of 0.5% per day assumed in the analysis of maximum credible accident."

b. Document Reference 56 indicates that the irradiated (5×10^7 R) cable can withstand a combined high temperature, steam and pressure environment (40 psig or 54.7 psia) for four days.

2.1.5 Test Specimen Aging

Document Reference 56 shows that the irradiated (5×10^7 R) cable can withstand a combined high-temperature and steam environment for four days at 40 psig and 142°C (287.6°F). Even though there is no evidence that the test samples were pre-aged prior to steam test, the test proved the capability to withstand 142°C for at least four days.

Based on the normal containment ambient temperature of 90°F (32.2°C), using the 10°C one-half life rule, the duration of aging required to prove 40-year life is seven days at 143°C. Even though the test duration was four days only against the requirement of seven days, it is our judgment that the combined effect of high-pressure, steam and high-temperature environment is far more severe than the anticipated in-containment environmental conditions following the worst case LOCA. The test pressure was 54.7 psia (40 psig) against the requirement of 41.7 psia and peak temperature during test was 289.6°F for four days against the requirement of 235°F maximum for one hour only.

Further, Document Reference 83 indicates that the maximum withstand temperature for SBR is 290°F which provides further credence to the conclusion that at relatively low temperatures such as 90°F (normal ambient) the cable chould be capable of performing satisfactorily for 40 years.

The cable pigtails which have been in service for approximately 18 years were recently visually examined. It was noted that the conductor insulation did not exhibit any sign of aging-related degradation such as cracks, brittleness, loss of flexibility, discoloration, etc, both inside and outside the containment. This is to be deemed as further evidence of aging qualification for the cable.

FRC EVALUATION:

FRC has reviewed the Licensee submittal, the referenced IEEE Transactions papers, and the manual for process engineering calculation excerpts provided by the Licensee as references. FRC has the following comments:

a. FRC is not aware of environmental testing conducted on the cable identified by the Licensee as this equipment item.

b. The Reference 1 information does not mention Buna S as a material evaluated as an insulation or tested in any of the tests conducted, as implied by the Licensee submittal, Section 2.1.2.

c. As stated elsewhere in this report on cables, the Guidelines require testing to demonstrate environmental qualification and identity of type between installed and tested equipment. Extrapolation of basic material information as attempted in the submittal on Big Rock Plant is not demonstration of qualification.

d. The summary reports on penetration tests do not identify any cable type, manufacturer, or material.

FRC CONCLUSION:

This equipment belongs in NRC Category V. Based on the information provided, FRC has no basis for determinating that the safety function can be carried out. This item should be either tested or replaced.

4.6.12 Equipment Item No. 53
Electrical Cables Located Inside Containment
Okonite, Model Not Stated
(Original Licensee References 9, 56, 1, and 83)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

1.0 Component: 3 kV power cable.

1.1 Single conductor #4/0 AWG nonshielded ozone-resisting oil-based rubber insulation, heavy-duty black neoprene jacket, Okonite cable.

2.0 The above cable is used in Class 1E penetration from inboard to outboard. This cable will not be exposed to containment spray or submergence.

2.1 The qualfication information of the subject cable is based on the following documents and discusssions:

2.1.1 Temperature

a. Document Reference 83, Figure D-19, shows the high-temperature limits of the various rubbers. The above referenced figure shows that natural rubber can withstand up to 270°F. It is an established fact that oil-based rubber has better heat withstand capability than natural rubber.

b. Document Reference 1 indicates that cable did not have any deteriorating effect when irradiated (5×10^7 R). Cable was subjected to high temperature of 287.6°F and steam environment for 25 days.

These values of 270°F and 287.6°F are higher than the 235°F required for Big Rock Point.

2.1.2 Humidity

Document Reference 56, Page 534, states that irradiated cable sample was subjected to conditions simulating the steam environment expected within the containment vessel. Water-filled jars containing the sample were maintained in a steam autoclave at 40 psig (142°C) for maximum period of 32 days. Table IX in Reference 1, IEEE Transactions states that irradiated cable (5×10^7 R) can withstand a combined high-temperature and steam environment for 25 days.

It is, therefore, concluded that the cable can function satisfactorily in 100% humidity environment.

2.1.3 Radiation

Document Reference 1 Table XI states threshold of gamma radiation damage (rad) for elastomer-based cable insulations. The table states that cable with 90°C oil-base insulation can withstand 10^8 rad gamma radiation.

It is concluded that the cable is qualified to Big Rock Point requirements based on the following:

a. Reference 56 indicates that the cable can withstand 10^8 rad gamma which exceeds the totals requirement of 7.3×10^5 gamma and also exceeds the total requirement of 1.373×10^7 R (1.3×10^7 R beta and 7.3×10^5 R gamma).

b. Cable jacket provides shielding against beta radiation. Therefore, the conductor insulation will be exposed to a significantly lower dose of radiation.

2.1.4 Pressure

It can be concluded that the cable is qualified to Big Rock Point requirement based on the following:

a. Document Reference 9 includes a report on final integrity test and leakage rate determination for reactor containment of the Big Rock Point Plant. This report indicates a satisfactory leakage rate of 0.21% per 24 hours at 10 psig and further concludes that the leakage at design pressure of 27 psig would be in the same order as the leakage at 10 psig. The report states "it is safe to say that the leakage at design pressure is far below the value of 0.5% per day assumed in the analysis of maximum credible accident."

b. Document Reference 56 indicates that irradiated (5×10^7 R) cable can withstand a combined high-temperature steam and pressure environment for 25 days at 40 psig (54.7 psia) and 142°C.

2.1.5 Test Specimen Aging

Based on the following evaluation, it is concluded that the cable is qualified for 40-years plus LOCA aging:

Document Reference 56 in Table II indicates the estimated life of oilbase insulation is 61 years at 70°C (168°F) which is far above the value specified for the Big Rock Point Plant.

FRC EVALUATION:

FRC has reviewed the Licensee submittal, IEEE Transactions, and the manual for process engineering calculation excerpts provided as references for this equipment and has the following comments:

a. The references provided as part of the submittal do not discuss oil base rubber, Okonite cable, or oil base rubber as a base material.

b. The Guidelines require that type testing on the equipment installed in the plant must be performed to demonstrate qualification of Class 1E electrical equipment. Extrapolation of data from related materials cannot be construed as providing any evidence of qualification or operability under the LOCA/HELB conditions.

c. The test summaries on penetrations do not mention the type or manufacturer of the cable.

d. FRC is not aware of testing on the cable because the description is not sufficient to relate it to known tests on cable manufactured by Okonite.

FRC CONCLUSION:

This equipment belongs in NRC Category V because there is no evidence of qualification or that the safety function will be performed reliably. This cable should be tested as installed or replaced.

4.6.13 Equipment Item No. 55
Cable Splices Located Inside Containment
AMP Model "Certiseal"
(Licensee References 4-36, 4-55, 4-59, and 4-61)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance, with design specifications, is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

Reference 4-36 is a discussion of calculations performed to establish radiation dose rate. Reference 4-61 applies to cable splices other than AMP, which are identified as Equipment Item No. 39.

Reference 4-59 is a memo of telecon on tensile strength and insulation resistance of splices.

These references do not establish, by either test or analysis of tests and materials, that the splices are capable of meeting LOCA requirements. It is suggested that the Licensee contact the manufac er to determine whether

testing or analysis has been conducted demonstrating the ability of the splices installed at Big Rock to meet the DOR Guidelines requirements on environmental qualification.

FRC CONCLUSION:

The Licensee did not address the item in the final submittal. If it is still installed, it belongs in NRC Category V.

4.6.14 Equipment Item No. 56
Junction Box Within Containment
Model Not Stated
(Licensee References 4-14 and 4-33)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

FRC has reviewed the Licensee referenced documents and has the following comments and conclusions:

a. The Licensee has cited as evidence of qualification for junction boxes located within containment

FRC would expect that the combination of the junction box and the item contained inside (terminal block/cable splice, etc.) should be the purpose of the qualification test rather than the junction box alone. In this regard, Reference 4-33 indicates that the Rumsey junction box houses State NT type terminal blocks. Reference 4-14 does not indicate what type terminal connector/block is inside the junction box.

b.

The Licensee should present evidence that the junction box and enclosed terminal connection device installed at the Big Rock Point plant

The effects of aging, radiation, and spray should also be addressed.

FRC CONCLUSION:

The Licensee did not address this item in the submittal. If still installed, it belongs in NRC Category V.

4.6.15 Equipment Item No. 57
Pressure Transmitters Located in Containment

Core Spray Pressure Transmitter (PT-186)
(Licensee Reference 4-28)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide environmental qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of a design basis events. A simple vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimen that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

Specific information relative to the transmitter's location and elevation inside containment was not provided, and it is possible (according to a reference document) that the instrument will be subject to submergence. The Licensee should review this item and offer qualification for submergence condition if it is possible for flooding to occur.

FRC has reviewed its files to determine if previous testing had been performed on the The following information is provided as guidance to the Licensee. A Westinghouse topical report, WCAP-8541, describes several qualification programs (seismic and environmental) for the Foxboro E10 series transmitters. Westinghouse states that, in all cases, the transmitters met the requirement of remaining operable for one-half hour after the accident. FRC's review of this reference has resulted in the following comments:

a. The information base not easily ascertained. Westinghouse Report No. WCAP-8541 contains descriptions of and results from the following qualification programs conducted for the by various test organizations:

   - Report No. [illegible]-6005 -- A LOCA exposure test was conducted (excluding radiation and chemical spray) on and model transmitters (10-50 mA dc). All units used the standard

   - Report No. T3-1013 -- A LOCA exposure test was conducted (excluding radiation) on model transmitters (4-20 mA dc). A junction box assembly was also tested. The units used amplifier part numbers

   - Report No. T3-1068 -- A radiation exposure test was conducted on and odel transmitters (4-20 mA dc and 10-50 mA dc). The units used amplifier part numbers N0148ND, N0148NL, and N0148PD. Failure of certain transmitters at high radiation levels was noted.

   - Report No. T3-1097 -- A radiation exposure test was conducted on improved amplifiers, modified because of the due to failures experienced during the previous test.

   - Report No. T4-6040 -- A dry oven bake, radiation, and hydrostatic test was conducted on box cover assemblies and associated 0-rings, and seals.

b. Report No. WCAP-8541 has presented the results of a variety of tests conducted on transmitters of varying models, amplifier part numbers, and accessories. The specific conclusions relative to qualification are obviously dependent upon the relationship between the test specimen and the actual installed equipment. The Licensee has identified the transmitter overall model numbers; however, many specific details with respect to transmitter identification have not been presented. 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 to those of the installed equipment. Therefore, an independent conclusion cannot be reached regarding the extent to which the two units were similar, and the validity of the test programs as evidence of qualification has not been established.

   In order to establish the relationship between the test specimen and the installed equipment, FRC concludes that the Licensee must provide the following additional information for the installed equipment:

- The full model number for all transmitters (for example,
- The transmitter case style (for example, A or B).
- The transmitter current output rating (for example, 4 to 20 mA dc or 10 to 50 mA dc).
- The transmitter top works amplifier part number (for example,
- The transmitter body material (for example, aluminum, iron, or stainless steel).
- The transmitter capsule assembly part number and O-ring part number (and material).
- The method of electrical connection and associated accessories (for example, Conax fitting and pressure seal junction box assembly).
- The transmitter special modification designation (for example, MCA/RRW).

c. The second LOCA test program (T3-1013) was more comprehensive than the first (Q9-6005). Various "Style B" transmitters with cast iron covers were tested. Westinghouse has stated that the greater heat sink provided by the cast iron cover should improve test results over the aluminum cover; however, a specific comparison of test results was not presented. FRC concludes that for the purpose of establishing qualification, the test program reported in T3-1013 can be considered the primary test.

d. It is not clear whether these transmitters will not become submerged, and therefore FRC concludes that the aspect of submergence should be addressed.

e. The Guidelines require that the test chamber temperature/pressure profile envelop the service conditions for a time duration equivalent to the period from the initiation of the accident until the service conditions return to normal values. Test Report No. T3-1013 has established that the test chamber temperature/pressure profile under all steam conditions, including chemical spray, exceeded the postulated accident profile, and therefore FRC concludes that this aspect of the qualification program is acceptable.

f. The Guidelines require that equipment operational modes during testing should be representative of the actual plant application requirements. In addition, failure criteria should include instrument accuracy requirements.

Test Report No. T3-1013 stated that the reference side of the sensing elements of the and transmitters was exposed to the test chamber environment, and that the pressure above atmospheric produced large negative-going output shifts. The adjusted output error for the unit ranged from -13% to -8% of span. The adjusted output error for the E11GM unit ranged from -8% to -3% of span. FRC concludes that this is presumably unacceptable, and reflects failure of the transmitter to perform with adequate stability. Unless the Licensee provides justification for acceptability of this maximum error range, FRC concludes that the unit has failed to qualify under environmental testing.

g. Test Report No. T3-1013 states that three connector and junction box assemblies were separately subjected to the same environmental test as the transmitters. The description of the test states that 3-XJB-I/25 MCA cast iron junction boxes and pressure seal assemblies (including blocks) were tested; however, no reference was made to The assembly performance was satisfactory. In Refererence 6, Westinghouse states that connectors used for electrical connection in this style transmitter were tested. These statements concerning the method of electrical connection employed on the tested transmitters are obviously contradictory. As stated previously, the field installation must be identical to the test setup. The test organization's report states that transmitter voltage supply and signal connections were made at the transmitter by splicing wires (separated by a bridge) and employing, and tape. This appears to have been accomplished (by observation of photographs in the test report) by splicing to 1-foot pigtail leads passing through a factory-sealed electrical fitting at the transmitter. The Licensee should provide the details of the method of electrical connection on the test specimens and on the units installed in the plant.

h. It is apparent that the referenced testing was conducted using transmitters that had been modified for environmental testing and designated as MCA (Maximum Credible Accident)/RRW (Radiation Resistant Wiring) units. The Licensee should verify that the installed units are so designated.

i. Test Report No. T3-1068 describes radiation testing conducted on the following transmitter models: (3 specimens) using 4-20 mA dc and amplifiers, (3 specimens) using 4-20 mA dc amplifiers, and (2 specimens) using 10-50 mA dc amplifiers. Two of the specimens were previously tested (T3-1013) in a steam-air-chemical-spray environment. These units were designated serial number (Model using a 4-20 mA dc amplifier) and (Model using a 4-20 mA dc amplifier). It should be noted that amplifier

was a prototype unit designed for nuclear service with radiation resistant wiring. Some discrepancies exist in the referenced report; the test organiztion report states that the amplifier for serial number (previously tested) was remote-mounted outside the radiation field, while the Foxboro summary of this report states that only the amplifier for serial number transmitter was located remote from the radiation source.

The summary conclusion of the test was that all units continued to function at a dose rate of 1 Mrad/hr to total doses of 76 Mrads or greater. However, failures did occur. The two 10-50 mA dc transmitters, Model (using the amplifier, failed: one unit's output went to > 0% at 76 Mrads, and then returned to half the normal output, the other unit's output went to half normal at 90 Mrads. Two of the 4-20 mA dc transmitters, Model (using the amplifier), continued to function with maximum errors of >3.75% until termination of the test. The unit with serial number (previously tested) operated with a maximum error shift of -3.3% up to a failure point of 86 Mrads. The transmitters using the amplifier operated with maximum error shifts of >4.85% up to 22 Mrads. The other unit ( ) exhibited possible failure for two hours at the 69 Mrad level of irradiation.

Failure of the amplifiers, both the 4-20 and the 10-50 mA models, was traced to a voltage drop across a type diode. This diode is used in all three amplifier models. Although failure occurred at radiation levels greater than the postulated accident levels for gamma radiation (20 Mrad), FRC concludes that degradation due to radiation did occur to units that were not simultaneously or subsequently exposed to a steam-air chemical spray environment. The Guidelines require that radiation exposure should be applied during the test sequence concurrent with, or prior to, the temperature and pressure/steam environment, if it is known that the device contains materials which can be degraded by irradiation. It has been established that the transmitters are susceptible to degradation by radiation exposure. The Licensee has not provided an analysis showing beta radiations can be disregarded for this equipment. In light of these considerations, FRC concludes that the test sequence for these devices should have included significant irradiation exposure prior to or concurrent with the temperature/pressure testing.

j. Test Report No. T3-1097 describes radiation testing conducted on amplifier assemblies only. It should be noted that a circuit modification, made because failures incurred during the previous test program (T3-1068), replaced diode type with type In addition, certain resistor and capacitors were replaced in the 10-50 mA dc amplifiers. Up to 22 Mrads, the amplifiers exhibited maximum shifts of -6% zero, +1% span, and -4.7% to -5.7% output. The

N0148TE(TJ) amplifier exhibited maximum shifts of >-2.5% zero, >+0.5% span, and >-2% output. The amplifiers experienced some difficulty. Two units functioned to 220 Mrads and one unit became erratic at 140 Mrads and then failed. Maximum shifts for the amplifiers were >-4.2% output, >4.2% zero, and >2.2% span. The failure was traced to a type transistor, which the report states is being analyzed.

Although the units were tested to radiation levels greater than the postulated accident level, FRC concludes that these specific amplifier assemblies have not been tested as an integral part of transmitters that have been exposed to a steam-air chemical spray environmental test. Therefore, comprehensive evidence of qualification has not been established.

k. Test Report No. T4-6040 describes hydrostatic leak tests conducted on eight transmitters having 316 stainless-steel cover assemblies. Four assemblies used part numbers the other four assemblies used propylene O-rings. All units were subjected to a dry oven bake exposure and a radiation exposure prior to hydrostatic testing. The results of the testing concluded that no appreciable leakage occurred. The report also states that the standard was exposed to the radiation and temperature environments and is therefore qualified. The Licensee should establish the specific correlation between this testing and the transmitters installed in the plant.

On the basis of the foregoing, FRC concludes:

1. The test report indicated that several different models of transmitters and amplifiers, using special modifications, were tested in a variety of combinations. However, the exact relationship between the installed transmitters and the appropriate test specimen has not been established. The Licensee should provide this detailed information (as indicated above in Item b).

2. The test reports indicate that transmitter models and are deficient with respect to adequate stability and accuracy. The Licensee should provide justification or additional information to show acceptability of the test results.

3. The Licensee should provide detailed information regarding the method of electrical connection at the transmitter for the test specimens and the installed units.

4. The test report indicated that the transmitters are degraded by radiation. The Licensee should provide evidence of radiation testing combined with a LOCA temperature/pressure exposure.

5. The Licensee should provide evidence that the improved radiation resistant amplifiers have been tested to a steam-air chemical spray environment as an integral part of an operating transmitter.

6. The Licensee should address the matter of qualified life.

LICENSEE RESPONSE:

[None]

FRC EVALUATION:

The Licensee has not provided a response to the DITER. In addition, this equipment has been deleted from the Licensee's most recent submittal, [90] and a justification for deletion was not provided. FRC notes that this transmitter is shown on drawing M-123 P & ID "Fire and Post Incident Core Spray System," the device monitors core spray pressure.

With respect to the transmitter, the Licensee has not provided additional references as evidence of qualification. Therefore, the specific deficiencies identified in the DITER remain unchanged.

FRC evaluated Reference 4-28 and concludes that this reference provides no information relevant to qualification. With respect to transmitter qualification, FRC concludes on the basis of the test report WCAP-8541:

- Evidence has not been provided to show that the improved radiation-resistant amplifiers have been tested to a steam-air, chemical-spray environment as on integral part of an operating transmitter.
- The Licensee has not addressed aging degradation and qualified life.
- The Licensee has not addressed the need for post-accident, long-term monitoring.
- The exact relationship between the installed transmitter and the appropriate test specimen has not been established. Various "E" series models were tested (such as and); however, the Licensee has stated that the installed model is a model The test specimen transmitters had a special modification designation MCA/RRW which denoted its use under severe environments.

- The test report indicates that the transmitters are deficient with respect to adequate stability and accuracy.

- The test chamber temperature/pressure profile under all steam conditions exceeded the required Big Rock Point accident profile for a time period of 2 hours; however, the test chamber profile did not envelop the required profile after the The test time duration, stated to be 24 hours, did not envelop the required accident profile 41-hour interval. In addition, after the the test chamber pressure was significantly lower than the required pressure (27 psig) for the remainder of the test.

- The Licensee has not provided information regarding the method of electrical connection at the transmitter for the test specimen and the installed units.

- The test report indicated that the transmitter is degraded by radiation. Evidence of radiation testing combined with a LOCA temperature/pressure exposure is therefore required.

FRC CONCLUSION:

This equipment item belongs in NRC Category V because of deficiencies in the test documentation that FRC has knowledge of. However, FRC concludes that there is a high likelihood of operability (12 hours) because of the comprehensive testing conducted on the transmitter. Short-term safety can therefore be accomplished. The Licensee has not provided evidence of qualification for this item. FRC concludes that the exact relationship to the appropriate test specimen has not been established, including special modifications for severe environments.

In addition, the specific application for this instrument is intermediate-term monitoring (30 days). Evidence of qualification for this component beyond 2 hours is lacking.

4.6.16 Equipment Item No. 58
Terminal Blocks Located Inside Containment

(Licensee References 4-14 and 4-33)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

FRC EVALUATION:

FRC has reviewed the referenced documents and notes the following:

a.

b. Reference 4-33 mentions a terminal block but identifies it as States type NT.

The referenced documentation does not appear to be relevant and, in addition, is confusing. FRC concludes that the Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequence of design basis events. A simple letter or vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing required that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been preaged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the components contains such materials, a qualified life for the component must be established.

It is suggested that the Licensee determine from the manufacturer whether any testing has been performed which would meet requirements of the DOR Guidelines for environmental qualification.

FRC CONCLUSION:

The Licensee did not address this equipment in the final submittal. If it is still installed, it belongs in NRC Category V.

4.6.17 Equipment Item No. 60
Cable Splices Located Inside Containment
Manufacturer and Model Not Stated
(Licensee Reference 4-61)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

FRC EVALUATION:

Reference 4-61 is an evaluation of waterproofing materials used to cover cable splices in containment.

The Licensee did not provide qualification test documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method for review for all Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A simple letter or vendor Certificate of Compliance with design specifications is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimens that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

FRC offers the following additional information:

a. In correspondence FRC reviewed in the equipment environmental qualification program, stated that in general some cable jackets may fail in LOCA conditions causing the splices to fail. The letter specifically identifies failures of EPR/Neoprene cable due to cracking during LOCA exposure.

b. Adhesives used in connection with heat shrinkable materials must also be compatible with the cables. Many adhesives are not satisfactory for radiation and water spray.

c. Waterproofing the outside of the splice will not change the radiation/cracking susceptibility of some cables and cable splices.

The Licensee should provide test and analysis information demonstrating that the splices meet the requirements of the DOR Guidelines.

FRC CONCLUSION:

The Licensee did not address this equipment in the final submittal. If still installed, the item belongs in NRC Category V.

4.6.18 Equipment Item No. 20
Pressure Switches Located Inside Containment
Static-O-Ring Model 9TA-54-11SSX12
Monitors Reactor Pressure
(Original Licensee References 4-23 through 4-25 and 4-62;
Final Licensee References 20, 43-45, 71, and 73)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee provided several reference documents to establish through evaluation techniques and associated test reports that the pressure switch was environmentally qualified. The following is a brief summary of these references and FRC's evaluation relating to the Licensee's overall evaluation.

Reference 4-24 consisted of internal correspondence within Consumers Power Company which demonstrated that a radiation evaluation had been conducted on sensitive materials used in the Static-O-Ring pressure switch. The evaluation demonstrated that the materials would not be likely to suffer any degradation within a 30-day period following a potential accident. The maximum dosage anticipated in the containment was 7 Mrad while the diaphragm material is capable of withstanding 10 Mrad.

Reference 4-23 is a manufacturer's certificate of compliance which provides pressure switch calibration data irrelevant to the environmental qualification documentation. It does confirm the diaphragm material as being a 316SS primary diaphragm with a Viton-O-Ring.

Reference 4-62 is an environmental test for pressure switch Model 26RZ-YY45-GM4X5 which is not the same model that is installed at the Big Rock Point plant. In addition, the environmental conditions were 260°F and 20 psig which is 7 psi lower than the pressure parameter designated for the Big Rock containment. The report was not complete; not all of its pages were submitted for review. The duration of the pressure/temperature test was not stated. The report stated that the O-ring between the rod and housing had conformed to the leading thread of the housing and was thin on one side. This could cause

an ineffective seal problem which presumably could be corrected by assuring a concentric chamber on the housing and employing a laminated O-ring. The Licensee did not state this was implemented for the Big Rock Point plant pressure switches. FRC therefore concludes that this reference is totally inadequate for demonstrating environmental qualification of the containment pressure switch.

Reference 4-25 consists of correspondence between the Licensee and the Static-O-Ring Pressure Switch Company, as well as a letter from Micro Switch, a division of Honeywell. It attempts to justify the MT-4R Micro Switch as having the ability to withstand 235°F temperature for 1 hour, 210°F for 1 hour, 195°F for 1 hour, and 180°F for 2.5 hours. Micro Switch stated that in its engineering opinion, the switch would not be affected; however, they recommended that testing be conducted to substantiate the opinion. In Static-O-Ring's opinion, the electrical switch would not reach the 235°F temperature but would stay in the 180°F range because of the mass of metal which surrounds the switch. FRC concludes that the device should have been tested to demonstrate operability under the specified pressure/temperature conditions for the time frame required by the plant profile curve.

In a subsequent transmittal dated October 21, 1980, CPC provided another test report (No. 71050) from Ogden Technology Laboratories, Inc. conducted on a Model 26R2-YV4J-CM4X5 pressure switch. The report stated that switch activation did not occur at its established point when the switch was subjected to a 260°F, 20 psig external pressure and allowed to stabilize for 30 minutes. Because this report does not describe a test performed on the Static-O-Ring model installed in the plant, a positive conclusion that the unit installed at Big Rock Point would also fail cannot be reached. It does, however, raise additional concerns regarding operability of this manufacturer's pressure switches.

None of the references addressed the submergence or chemical spray failure mechanisms; however, both conditions should be addressed due to the pressure switch locations in the plant. In addition, the Licensee has stated that half of the switches will be subjected to submergence. Since the pressure switch contacts "open" to activate the reactor core spray motor

operated valve to the open position, the fact that qualification documentation is lacking for submergence or spray could result in common-mode failure due to water shorting out of the contacts and subsequently preventing the valves from traveling to the open position. FRC concludes that this potential concern should be addressed by either testing or reference to a previous prototype test on this specific model switch.

In summary, all of the references combined do not provide sufficient evidence that the pressure switch has the necessary qualification or statements regarding aging and qualified life. The Licensee should provide additional substantiating documentation.

LICENSEE RESPONSE:

Component Static-O-Ring Pressure Switch PS-IG11A-H

These pressure switches are used to provide a core spray valve open signal when reactor pressure is < 200 psig. In the electrical scheme, these switches have contacts in series with the reactor water level switches (LS-RE09). Therefore, both these pressure switches and the reactor water level switches must remain operable for the same length of time. The time required was established at 1 hour. The basis for this time is that it is the longest time for actuation of the core spray system when the LOCA is a small steam line break (0.008 ft^2). For the 0.008 ft^2 break, the reactor depressurizing system (RDS) must first operate to reduce the reactor pressure to < 200 psig. The RDS will operate when low reactor water level is reached in this small steam line break senario.

Per the Loss of Coolant, Emergency Procedure EMP 3.3, the operator is directed to manually initiate the RDS if he deems it necessary. With the loss of steam drum level indication and alarm as well as all of the other symptomatic indications, it is likely that the operator would take the required action early in the accident rather than wait an hour for automatic action. Therefore, in reality, the instruments would not have to endure an hour in the environment caused by a small steam break. Further, the small steam line break produces a less severe environment than the large LOCAs until the RDS functions. The larger-sized LOCAs allow the instruments to perform their functions early in the events, thus significantly reducing the lifetime required to demonstrate qualification.

The actual atmosphere conditions under which these switches will operate are pressure less than 39 psia and temperature less than 235°F.

These switches were environmentally qualified in a test. During the test, the pressure switches were placed in a test chamber and nitrogen was added. The conditions were raised to 260°F and 20 psig and held for 2 hours. This is of a longer duration than required. The switches were also test-operated during this time.

The test temperature was greater than that needed for qualification but the pressure was 7 psig lower. The difference in pressure is not significant because the switches are in an explosion-proof housing as compared to a NEMA 4 housing used on the test specimen. The additional pressure will have no effect on the explosion-proof housing.

Steam and spray were not used in the test; however, the sealed explosion-proof housing will prevent this from having any significant effect on the switches. Therefore, the lack of use of steam or sprays does not invalidate the test.

There were differences between the test specimen and the installed switch but these differences were not significant and do not invalidate the test. (1) The pressure range was different. (2) The test specimen had a weathertight housing vs a sealed explosion-proof housing for the installed specimen. (3) The microswitch used in the installed specimen was evaluated by the manufacturer to be able to withstand the environment (Ref. 73). (4) The tested pressure switch had 5 options which were not listed. The installed transmitter has 12 options. The 12 options make the switch more insensitive to the environment (Viton O-ring cover gasket, silicone diaphragm, high-temperature shaft, 316 SS primary diaphragm, Viton O-ring) and also include items like certification, data sheets, cleaning and packaging. It is therefore considered that the options in the tested sample which may not be in the installed switch are not important.

The switch's ability to withstand radiation was evaluated. It was found that even the most sensitive component can withstand greater than 10^6 rads.

The pressure switches actuate early in the event and the LOCA event is expected to thermally age the components very little. The normal aging environment is a temperature averaging 70°F. Since these switches are on a periodic surveillance for calibration, a failure could be detected. No failures have been found. It is concluded that (1) with the uncertainty in determining the lifetimes for age sensitive materials (see Section IV), (2) with the low ambient temperatures experienced by the instruments, (3) with the short time needed during an accident and (4) with the known good history of the instruments during calibration intervals. These instruments are acceptable for continued operation.

The switches are encased in a sealed explosionproof housing. This results in low temperature increases to the switch intervals prior to actuation.

The pressure switches are qualified for the environment they will experience. They were tested for high temperature and pressure for a duration longer than the time required for operation in the event of an accident. The test did not include sprays or humidity; however, the switches are sealed in an explosionproof housing and will not be affected by spray or humidity. They were evaluated and found able to withstand the radiation and aging.

FRC EVALUATION:

From the reference material made available to FRC and the evaluation performed by the Licensee, the following deductions can be made:

1. According to the Licensee's response, differences between the test specimen and the installed switch were not insignificant and do not invalidate the test. Also, the options in the tested sample which may not be in the installed switch are not important.

2. According to Reference 71, the pressure switch did not properly actuate during the environmental test at 260°F and 20 psig. This is 25°F higher than the plant-specific temperature of 235°F and a satisfactory re-test apparently was not conducted.

3. The deduction can be made, therefore, that the unit in the plant would fail the test as did the test specimen, although the Licensee response stated that the switches were environmentally qualified in the test.

In addition, the test time period was not equal to the 1 hour plus time period for the switch to perform its intended safety function because of the test specimen's failure within a 30- minute stabilization period. [71]

The problem pointed out in the DITER concerning submergence qualification has apparently been addressed with the change to a Viton O-ring cover gasket. The Licensee should confirm that the O-ring resolved the submergence issue.

FRC CONCLUSION:

This level switch belongs in NRC Category V because it references a type test claimed by the Licensee to be equivalent to the switch in the plant which failed its qualification test at a temperature 25°F higher than the plant-specific temperature and in a shorter test period. Justification for interim operation has not been provided.

4.6.19 Equipment Item No. 21
Pressure Transmitters Located in the Electrical Penetration Room

Containment Pressure Indication and Vacuum Relief (PT-173, -174, and -187)
(Original Licensee Reference 4-32; Final Licensee Reference 46)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide qualification documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment required to mitigate the consequences of design basis events. A simple vendor Certificate of Compliance with design specifications or catalog information is not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated enviroment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test specimen that had not been preaged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life of the component must be established.

Although the transmitter's environment has been listed as harsh, the maximum temperature and pressure are only 100°F and 15 psia, respectively. The relative humidity is expected to climb to 100% and is not expected to become a failure mechanism since the transmitter is enclosed in its own housing and the high energy line break outside containment is anticipated to be of short duration.

FRC has reviewed its files to determine if any previous testing has been conducted and has been unable to find any test reports for the differential pressure transmitter. This search does not, however, rule out the possibility that the unit has undergone qualification testing. The Licensee should contact the manufacturer to determine its availability.

LICENSEE RESPONSE:

These pressure transmitters are used to indicate containment pressure and, in the case of PT-173 and PT-187, actuate valves for vacuum relief. The vacuum relief function, if needed, will be required up to 12 hours after the LOCA. The transmitters are located in the electrical penetration room. The room experiences a temperature rise during the LOCA as it is adjacent to the containment (spherical steel structure). To determine the temperature in this room, a CONTEMPT run was performed for the double-ended rupture of the main steam line. This results in the highest containment temperature. One of the assumptions was that the containment is completely insulated and, therefore, no heat is transferred out of containment. This is a very conservative assumption. The peak inside containment wall temperature was found to be 182°F. The room temperature during the accident will be lower than this, due to the conservative assumption of zero heat transfer out of containment and the time lag to transfer heat through the shell and through the room to the transmitters (located at least 6 feet away). The transmitters are able to operate in ambient conditions of 220°F and, therefore, are qualified for temperature.

The radiation dose, which these transmitters will obtain, is calculated to be 5×10^4 rads in 30 days. The containment pressure is back to atmospheric in approximately 24 hours. The dose in 24 hours is 3×10^4 rads (ratio of containment atmosphere dose at 24 hours to that at 30 days). The sensitive parts of the transmitter are housed in an explosion proof casing and are also shielded by other components. The dose is, therefore, even lower. All commonly used materials can withstand this radiation level and it is concluded that the trasmitter is qualified for radiation.

No data on aging could be located. The transmitters are in a normal environment which ranges from 40°F to 100°F (seasonal variation). CP Co will continue to search for age-sensitive materials and, if found, will be replaced by June 1982. Continued operation is justified by the arguments in Section IV.

FRC EVALUATION:

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

Reference 46 is a product data sheet, and does not consititute qualification documentation.

FRC notes that the adverse environmental condition stated by the Licensee for this component is 0.052 Mrad and 182°F, and the required operating time is 30 days.

FRC concludes that this component is lacking documentation reflecting operability under the specified environmental service conditions. The Licensee has provided some justification for interim operation. The Licensee stated that age-sensitive materials in the equipment will be replaced if found. However, FRC notes that the Licensee has not provided a definitive statement regarding equipment replacement or modification.

FRC CONCLUSION:

The equipment item belongs in NRC Category V because the equipment is lacking qualification documentation; however, the Licensee stated that the equipment will be replaced if age-sensitive materials are found. FRC notes that the Licensee has not reached a definitive conclusion as to disposition of this item in accordance with NRC requirements.

4.6.20 Equipment Item Nos. 61A & 61B
Terminal Blocks and Splices Located Inside Containment
A: Stanwick Co. Terminal Block, Model Not Stated
B: Splice, Manufacturer Not Stated
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

Reference

FRC EVALUATION:

On December 30, 1980, Consumers Power Company issued Licensee Event Report (LER) No. 80-048 [109] which reported on the degradation of some cable splices and terminal blocks which were located inside the containment. These items have a safety function because they are connected to wiring which controls devices in the emergency core cooling system.

The items referenced in this LER were not initially identified by the Licensee in the initial submittal, were not incorporated into the DITER, and similarly were not identified in the last Licensee submittal [90]. For these reasons they have been included as a special equipment item in NRC Category V.

The Licensee stated that these items would be replaced by terminal blocks made by States Co., and the splices would be replaced by Raychem Co. shrink fit sleeves.

In the period prior to the issuance of the LER, FRC had reviewed the existing Big Rock Point plant's States Co. terminal blocks (Equipment Item 38B) and found that sufficient evidence did not exist to support their qualification as per the DOR Guidelines requirements. Therefore, the terminal blocks which are being substituted have questionable qualification documentation and need further Licensee investigation so that qualification can be adequately demonstrated.

In regard to the Raychem splice system, the Raychem splices have been qualified to specific plant conditions but the cables to which the splices are joined must also be qualified or the splices could experience failure.

FRC CONCLUSION:

The terminal board and splices which were initially installed at the Big Rock Point plant have been determined by the Licensee as unqualified and belong in NRC Category V. In addition, the replacement terminal board may not have qualification established; further investigation by the Licensee is necessary.

4.7 NRC Category VI
EQUIPMENT FOR WHICH QUALIFICATION IS DEFERRED

This section includes equipment items which have been addressed by the Licensee in the equipment environmental qualification submittal but for which 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. 6
Level Switches Located Inside Containment
Jo-Bell Model Type R
Monitors Containment Water Level
(Original Licensee Reference 4-2; Final Licensee References 18, 19, 64, 65, and 81)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee, in Reference 4-2, has provided an evaluation of the Jo-Bell level switch to demonstrate qualification. The Licensee pointed out that the only non-steel parts are the switch and an asbestos gasket. The temperature rating of the switch is 250°F and the pressure rating of the device is 25 psig. Although the design temperature rating is satisfactory, the maximum working pressure is 2 psi lower than the containment design pressure and does not therefore envelop the pressure service condition. The Licensee mentioned that the only item susceptible to radiation was the asbestos gasket.

Two of the four level switches are subjected to submergence conditions; however, no evaluation or testing was conducted to demonstrate operability during submerged conditions.

The Licensee did not provide either a detailed drawing or materials list which could accurately define this component's basic design features or items susceptible to thermal or radiation degradation. Aging of the device was not addressed nor was a qualified life established.

FRC concludes that a prototype unit should be pressure and submergence tested in order to demonstrate that the unit will not fail under these conditions. A more detailed evaluation is necessary to assure that all materials have been reviewed for thermal and radiation aging mechanisms.

LICENSEE RESPONSE:

Component Level Switches LS-3562 through LS-3565

Level switches LS-3562 through LS-3565 are used to provide containment water level indication. These four switches are located at elevations 574', 579', 587'and 595'. The switches close at contact when the water level reaches them. The electrical scheme is such that the switches are wired in parallel to the power supply. Failure of one switch will not impair the rest of the circuit.

The internal switch and trip mechanism, rated to 160°F, are housed in an explosion-proof steel housing. The containment water temperature is above 160°F for, at most, 5.5 hours. The atmosphere temperature is above 160°F for a longer period of time. The housing will cause the temperature of the internal switch to lag that of containment such that the switch temperature is above 160°F for a shorter time. The switch is all metal construction except for an asbestos gasket and neoprene wires. The switch is rated at 44.7 psia, which is in excess of the maximum containment pressure. The switch is sealed with the asbestos gasket while the electrical connections are sealed with Chico sealing compound placed in a sealing cage. It should be noted that these switches were designed to be submerged. Therefore, it is not expected that sprays, humidity or submergence will affect these switches. The known organic materials, asbestos and neoprene are acceptable for service to a radiation dose of at least 5×10^6 rads, whereas the 24-hour gamma dose in water is 1.1×10^6 rads. Further analyses of these materials may be required for determination of their sensitivity to thermal aging as was discussed in Section IV of this report.

In summary, no type test of this equipment is available as required by the DOR Guidelines.

The reader should note that these switches may be considered as backup indication to LT-3171. It is currently planned to install a new level indicating system in the fall of 1981.

FRC EVALUATION:

The Jo-Bell level switches have not been qualified to DOR Guideline requirements for submergence, and no evidence was presented by the Licensee that would demonstrate that the switch was designed for 44.7 psia as indicated by the Licensee. Reference 19 stated that the 44.7 psia maximum design pressure value applied only to level switches made after 1961. The Licensee did not confirm which switches were made after 1961 or which switch was furnished to the Big Rock plant. The level switch at the 574' elevation would

be exposed to a potential hydrostatic flood level head pressure of approximately 7 psi in addition to the containment's air/steam pressure of 41.7 psia. Therefore, the switch could be exposed to a 48.7 psia environmental service condition.

Although an analysis was provided to demonstrate that radiation aging was not a problem, the Licensee failed to provide a statement on qualified life.

The Licensee has stated that plans are under way to install a new level indicating system in the fall of 1981.

FRC CONCLUSION:

This level switch belongs in NRC Category VI because it is deferred as a TMI action plan item in accordance with Section 2.2.5. Justification for interim operation exists because this type of level switch was designed for submerged conditions, although no testing has been indicated.

4.7.2 Equipment Item No. 16
Motor Starter Located in the Post-Incident Room
General Electric Model CR109
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

NONE

LICENSEE RESPONSE:

This motor starter is for MO-7066, the cooling water valve to the core spray heat exchanger. The valve is actuated manually in the control room when the containment sump recirculation mode begins. The time required to reach recirculation level in the sump is a maximum of 21 hours. During this time, the core spray heat exchanger room remains at ambient conditions with respect to temperature, humidity, pressure and radiation. Temperature and radiation following the start of the recirculation are increased due to the temperature and radiation level of the sump water and temperature from the core spray pump motor heat. After the valve is required to open to allow cooling water to the heat exchanger, it will remain open for the duration of the event. If the valve actuator of the subject starter fails to operate to open the valve, it or its bypass can be opened manually in the core spray room. The time to begin recirculation is from 4 to 21 hours. No feasible failure of the starter can be determined that would cause the valve to reclose once it has been opened.

Because the starter operates when the room is at ambient conditions, and failure prior to this can be overridden by manual action, the starter is considered acceptable.

FRC EVALUATION:

This equipment item was not provided by the Licensee for review in the original submittal. The Licensee has stated that this equipment is located in a mild environment before it is required to operate. FRC therefore concludes that the qualification review can be deferred until after February 1, 1981 in accordance with NRC criteria presented in Section 2.2.3 of this report.

The Licensee did not reference any qualification documents. Because the starter is providing an important safety-related function, qualification documentation is necessary and the Licensee should provide additional information to show the possible failure of the starter and the electrical circuit to which it is associated. In addition, the Licensee needs to determine the qualified life of the starter.

FRC CONCLUSION:

This equipment item belongs in NRC Category VI based on qualification deferment in accordance with criteria stated in Section 2.2.3 of this report. The Licensee should provide information that would depict the actual starter's components and investigate different modes of failure.

4.7.3 Equipment Item No. 27
Solenoid Valves Located in the Turbine Building
ASCO Model HTX-830061RF
Actuates Isolation Valves (SV-4895, -4896, -4922)
(Licensee Reference 49)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

These solenoid valves are automatically de-energized by the safety system on an isolation trip from either the containment high-pressure or low reactor water level signals. The valves are supplied with

redundant valves inside containment (SV-4869, 4891 and 4876, with their respective control valves). Following de-energization, the valves will remain in the same state throughout the post-incident period. Failure of the elastomers in the solenoid valve will result in the control valve remaining in its fail-safe position; i.e., closed. Automatic de-energization of the valve on an isolation signal will occur well within one hour (see SV-4876, 4869, 4891) after a break; and, because the valve is located outside containment, it will experience no harsh environmental changes in this time. Temperature, pressure and relative humidity will remain at ambient conditions outside of containment. The total integrated 30-day radiation exposure at the containment surface is 76 krads, which is less than the threshold limit of greater than 1 Mrad for the valve elastomers.

The aging of the elastomers is not significantly accelerated during the post-LOCA period except, due to the higher than normal radiation which is shown above, the 30-day exposure is less than the threshold limit. The valves are not currently on a preventative maintenance program to replace the aged and worn components. In order to provide aging qualification, the valves will have the age-susceptible components replaced by June 30, 1982.

In summary, the valves are not required for HELB outside containment. Their environment, except for radiation, is not harsh and the materials are resistant to the accident dose received. To provide qualification for aging, a preventative maintenance replacement program will be initiated and maintained on a 5-year cycle.

FRC EVALUATION:

The Licensee has stated that this equipment is located in a mild environment except for radiation exposure. Considering that operation occurs early in the accident scenario, the 1-hour dose would be quite small. FRC concludes that the qualification review can be deferred until after February 1, 1981 in accordance with NRC criteria presented in Section 2.2.3 of this report.

The Licensee's statement that the preventive maintenance program will be initiated by June 30, 1982 is a cause for concern. The nonmetallic materials used in the solenoid degrade as a result of the relatively high internal temperature caused by constant energizing. The preventive maintenance program (PM) should commence immediately.

FRC CONCLUSION:

This equipment item belongs in NRC Category VI based on the EEQ review deferment described in Section 2.2.3. It is recommended that the PM program begin immediately.

4.7.4 Equipment Item No. 28
Solenoid Valve Located in the Turbine Building
ASCO Model 830060R
Actuates Isolation Valve (SV-4897)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

This solenoid valve isolates demineralizer water to the containment building via manual operation from the control room. It is backed up by an isolation check valve which is located inside containment. The control valve closes when the solenoid valve is de-energized. Based on discussions with ASCO, in all probability when in the de-energized position, failure of the solenoid valve elastomers will result in the valve remaining in the fail-safe mode. Since the valve must be manually operated to close following a LOCA, the emergency procedure will have to be modified to identify this action to the operator. This manual operation will be accomplished within one hour after the event. After closure the valve will remain in its de-energized position for 30 days.

Temperature, pressure, and relative humidity all remain at ambient conditions outside containment. The total integrated 30-day radiation exposure at the containment surface is 76 krads which is less than the threshold limit of greater than 1 Mrads for the valve elastomer BUNA N.

The aging of the elastomers is not significantly accelerated during the post-LOCA period. Replacement of aged and/or worn components on a periodic basis will be done to maintain a qualified life for the valve. The program will begin by June 30, 1982.

Based on the normal environment it encounters with respect to temperature, pressure and humidity, the ability to withstand accident radiation, and because it is only necessary to de-energize the valve following a LOCA, the solenoid can be assumed qualified. Aging affects the nonmetallic components but they will be replaced. The backup check valve is leak tested each refueling outage and will maintain the containment boundary, and it also provides an acceptable reason to maintain operation. With the preventative maintenance replacement program the valve will be qualified.

FRC EVALUATION:

The Licensee has stated that this equipment is located in a mild environment except for radiation exposure. Considering that operation occurs early in the accident scenario, the 1-hour dose would be quite small. FRC concludes that the qualification review can be deferred until after February 1, 1981 in accordance with NRC criteria presented in Section 2.2.3 of this report.

The Licensee's statement that the preventive maintenance program will be initiated by June 30, 1982 is a cause for concern. The nonmetallic materials used in the solenoid degrade as a result of the relatively high internal temperature caused by constant energizing. The preventive maintenance program should commence immediately.

FRC CONCLUSION:

This equipment item belongs in NRC Category VI based on the EEQ review deferment described in Section 2.2.3. It is recommended that the PM program begin immediately.

4.7.5 Equipment Item Nos. 32A and 32B
Solenoid Valves Located in the Sphere Ventilation Room
32A: ASCO Model 8316C44
32B: ASCO Model FT-8316D44 Actuates Containment Ventilation Valves (SV-9151 Through 9154)
(Licensee Reference 49)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

These valves are the pilot valves for the containment ventilation supply and exhaust valves. They close on all scrams and containment high radiation. They may be required to reopen in the event of a containment vacuum condition. The valves are located outside containment and are subjected to normal ambient temperature, pressure and relative humidity following a LOCA. Radiation during a 30-day explosure will be 0.19 Mrads which is less than the 7.0 Mrads for a 25% damage valve of the BUNA N

elastomers. The valves presently undergo a preventative maintenance program to replace the valve elastomers. A change out is scheduled for the upcoming 1980 refueling outage. Based on the valves being located in a normal environment, both before and after a LOCA, and the present PM schedule, the valves are considered qualified for a 40-year plus LOCA lifetime.

FRC EVALUATION:

The Licensee has stated that this equipment is located in a mild environment except for radiation exposure. FRC concludes that the qualification review can be deferred until after February 1, 1981, in accordance with NRC criteria presented in Section 2.2.3 of this report.

FRC CONCLUSION:

This equipment item belongs in NRC Category VI based on qualification deferment in accordance with criteria stated in Section 2.2.3 of this report.

4.7.6 Equipment Item No. 33
Solenoid Valves Located in the Sphere Ventilation Room
ASCO Model WP-80306
Actuates Isolation Valves (SV-9155, 9156)
(Licensee Reference 49)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

These solenoid valves are the ventilation probe valves. They close on all scrams. The valves are in series and are located outside containment. They will de-energize and close, in their fail-safe position, on a scram signal well within one hour from a break and remain closed throughout the post-LOCA period. The valves are direct acting solenoid-operated valves which provide containment boundary when closed. Containment pressure aids the seating of the valves. Following a break the valves will remain in normal atmosphere with respect to temperature pressure and relative humidity. Radiation during the 30-day post-LOCA period will be 0.19 Mrads which is less than the 7 Mrads determined to be the 25% damage valve for the BUNA N seating material. These valves will be put into the preventative maintenance program to change out the valve elastomers. Local leak rates testing of these valves has shown no apparent deterioration of the valve seats. Because the valves operate in

a normal environment, will not be subject to damaging radiation and presently provide an adequate containment boundary, the valves are acceptable as is. The preventative maintenance program will be initiated by June 30, 1982 to replace the age-susceptible materials.

FRC EVALUATION:

The Licensee has stated that this equipment is located in a mild environment except for radiation exposure. Considering that operation occurs early in the accident scenario, the 1-hour dose would be quite small. FRC concludes that the qualification review can be deferred until after February 1, 1981 in accordance with NRC criteria presented in Section 2.2.3 of this report.

The Licensee's statement that the preventive maintenance program will be initiated by June 30, 1982 is a cause for concern. The nonmetallic materials used in the solenoid degrade as a result of the relatively high internal temperature caused by constant energizing. The preventive maintenance program should commence immediately.

FRC CONCLUSION:

This equipment item belongs in NRC Category VI based on the EEQ review deferment described in Section 2.2.3. It is recommended that the PM program begin immediately.

4.7.7 Equipment Item Nos. 18A and 18B
Pressure Switch Located in the Core Spray Room
A1: ITT-Barton Model 289A, Strainer High d/P Alarm Function
B1: Mercoid Model DAW-23-153, Core Spray Pump Discharge Pressure
(Original Licensee Reference 4-39, 10)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee did not provide qualification documentation for this equipment as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E electrical equipment which is required to mitigate the consequences of design basis events. A

simple vendor Certificate of Compliance, with design specifications, or catalog information, are not considered adequate or sufficient. Specifically, qualification by type testing requires that the simulated enviroment in the test chamber envelop the specific service conditions identified. In addition, tests which were successful using test speciman that had not been pre-aged may be considered acceptable provided the component does not contain materials known to be susceptible to significant degradation due to thermal and radiation aging. If the component contains such materials, a qualified life for the component must be established.

In Reference 10, the Licensee stated that the differential pressure switch is exposed to a harsh environment in the core spray room only after the core spray heat exchanger is placed in operation and after the differential pressure switch has performed its intended fuction. In addition, the worst-case enviroment will be one of high radiation since HELBs or submergence will not be a problem. FRC's review of the Process and Instrument Diagram M-123 depicts the differential pressure switch as providing an alarm function only, and its failure would therefore not degrade any safety-related equipment.

FRC concludes that environmental qualification is not required for this instrument because of its location in a basically mild environment during the period when the device must function. This function has a high likelihood of being performed prior to the occurrence of any long-term degradation.

LICENSEE RESPONSE:

Component Pressure Switches PS-638 and PDIS-7814

Both of these switches are in the core spray room. This room is not subjected to harsh environments from the LOCA until the containment sump is filled and the operator switches to the recirculation mode for core cooling water. Pressure switch PS-638 senses pressure on the discharge header from the core spray pumps. If the pressure is < 50 psig, an alarm is provided in the control room. Its primary function is to inform the operator of the failure of a core spray pump to start and therefore is not required subsequent to the start of the recirculation mode. Core spray flow indication is also provided by FT-2162 and FT-2163.

Differential pressure switch PDIS-7814 measures the dp across a basket strainer that filters firewater to the core sprays. This is an emergency

path for the core spray water addition, and would not normally be used (Reference MO-7072). Until recirculation, the environment is normal. The operator is directed by procedure to align the basket strainer to the "open basket" position prior to going to the recirculation mode. In the "open basket" position, PDIS-7814 serves no purpose.

Thermal aging is not considered to affect LOCA qualification of these devices for the following reasons: (1) the normal temperature is 70°F, a low temperature in terms of significant thermal aging, (2) the function of these devices is completed prior to being subjected to a harsh environment and (3) the switches have backup instruments that provide the necessary information, that is FT-2162 and FT-2163.

FRC EVALUATION:

FRC concurs with the Licensee's evaluation of the role that these pressure switches play in the overall plant operation. For the accident condition and time period for which they are required to function, the switches are located in a mild environment.

FRC CONCLUSION:

This equipment belongs in NRC Category VI. Since the equipment item is located in an area where the environmental service conditions are non-harsh, the qualification review of this equipment has been deferred by the NRC in accordance with criteria presented in Section 2.2.3 of this report.

4.7.8 Equipment Item No. 23
Radiation Monitor Located in the Electrical Penetration Room
Technical Assoc. Model CP-MU
(Licensee References 36, 43, and 48)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE RESPONSE:

Component Survey Meter - Core Damage Monitor

The survey meter is located in the electrical penetration room. Its function is to detect radiation inside containment and provide this indication to the operator. The meter is not safety-related and plant shutdown can be achieved without its use. The probe consists of an ionization chamber which is encased in an aluminum housing. The unit is designed to operate in high humidity. The electronics are in a separate

room which is a "mild" area. The cable used to connect the probe to the electronics is jacketed and has polyethylene insulation.

The meter was installed late in 1979 and is temporary until the high-range monitors are installed as part of the TMI-related modifications. The probe itself is an ion chamber. Due to its construction, it can withstand 182°F. The probe is qualified for aging as it is brand new and the temperatures in the room over the 30 days will not exceed 182°F. The temperatures in the room will most likely not even reach 182°F due to thermal inertia in the containment shell, the room atmosphere and the housing and cable jacket. The monitor is qualified for the temperature and radiation as the electronics are placed in another room. The temperature and radiation levels of 50 krads will not affect the aluminum housing or polyethylene insulation (1×10^8). Therefore, the monitor is qualified.

FRC EVALUATION:

The EEQ review is deferred until after February 1, 1981, as part of the TMI Action Plan outlined in Section 2.2.5.

FRC CONCLUSION:

This level switch belongs in NRC Category VI because it can be deferred as part of the TMI Action Plan, as discussed in Section 2.2.5.

5. CONCLUSIONS

FRC's review of the qualification documentation and other information submitted by Consumers Power Company for the safety-related electrical equipment items and their applications in the Big Rock Point Plant has resulted in the following categorizations:

NRC Category	Compliance With The DOR Guideline Requirements	Quantity Of Equipment Items
I.a	Equipment Fully Satisfies All Applicable Requirements for Life of the Plant	0
I.b	Equipment Does Not Meet All Applicable Requirements for Life of the Plant; However, Deviations are Judged Acceptable	0
II.a	Equipment Fully Satisfies All Applicable Requirements for Less Than Plant Life	7
II.b	Equipment Fully Satisfies All Applicable Requirements Provided That Specific Modifications are Made	0
II.c	Equipment For Which the Licensee Has Not Provided Evidence of Full Compliance with the Guidelines, but Which is Judged by FRC To Be Satisfactory for Less Than Plant Life Based on Total EEQ Program Review	1
III	Equipment is Exempt from Qualification	2
IV.a	Equipment has Qualification Testing Scheduled	0
IV.b	Equipment has High Likelihood of Operability; However, Proper Qualification Documentation is Not Available	25
V	Equipment is Unqualified	24
VI	Equipment Qualification Review Is Deferred	11

The following conclusions represent a summary of the results of the equipment environmental qualification (EEQ) evaluation conducted by FRC in accordance with the methodology presented in Section 3. The major deficiencies that have been identified are shown in the Qualification Summary Review Sheets at the end of this section.

The evaluations are based on the available qualification documentation provided by the Licensee, complemented in several cases by FRC's familiarity with other relevant technical information. The results of the review do not necessarily imply that all of the equipment is unreliable, is unsafe, or represents a significant plant safety issue. They do, however, point out that for many equipment items qualification documentation 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 to assure that safety-related equipment that exhibits age-related degradation be identified and, if necessary, replaced. No summarized equipment maintenance records have been submitted to FRC to demonstrate that either (i) a specific equipment item's maintenance activity has been random or routine, or (ii) some systematic degradation has occurred, requiring further analysis and replacement consideration.

The Licensee has offered several system-related arguments to justify continued operation and/or to exempt certain equipment items from qualification review: (1) 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 I.

The present review of the safety-related electrical equipment for the Big Rock Point Plant involved only the investigation of the equipment located in the "harsh environment" areas (i.e., containment drywell, electrical penetration room, core spray room and pipe tunnel) and needed to ensure hot shutdown of the plant. The EEQ review of equipment items located in "mild"

areas and those which are needed to bring the plant to a cold shutdown condition have been deferred by the Licensee until after February 1, 1981.

The Licensee is currently reviewing its containment temperature/pressure profile because a preliminary investigation into a potential MSLB occurrence had postulated excessive containment temperatures. If these preliminary results are found to be reasonably accurate, then the electrical equipment which is located in the containment would have to be reevaluated for a significantly higher temperature.

Presently the Licensee is attempting to correct a terminal board and cable splice qualification problem which was reported by the Licensee in LER 80-048 on December 30, 1980 as discussed in Section 4.6.19. The Licensee should ensure that proper qualification documenation exists for the replacement items and submit the information to the NRC for review.

FRC has noted in the review of the Licensee's comments regarding the component cooling water pump motors that their failure during a postulated HELB occurence has been assumed and that the backup service water system required air-operated valves to be operable. The Licensee did not include the instrument air system in the list of safety systems needed for the plant's hot shutdown. A review of this system's safety-related interface with the component cooling water and service water systems needs to be conducted by the Licensee.

Overall, the Licensee's submittal was well organized and demonstrated that the Licensee had programs in place to qualify equipment and determine environmental conditions. Some of the details for qualification and a few of the environmental parameters verifications are still in progress and need to be completed when the "mild area" review is conducted.

FRC concludes that most of the safety-related items cited by the Licensee do not have the qualification documentation necessary to unquestionably demonstrate that the equipment can withstand the stated environmental service conditions. Several equipment items probably could have been qualified if the Licensee had provided more detailed analyses and justifications for qualification.

The Equipment Environmental Qualification Summary Form presented on the following pages shows the overall results of the FRC review. The entries on this form delineate the overall status of each specific equipment item with respect to compliance with the Guidelines criteria defined in Section 2 and the resultant qualification categories 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 categorized as unqualified or qualification not established.

L = A limiting condition with respect to qualification in that the qualified life has not been established by the Licensee.

O = NRC qualification category.

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

* = The Licensee stated the following:

EQUIPMENT WORK SHEETS AND SUMMARIES

NOTE: Although it is stated that certain components will be replaced, Consumers Power Company reserves the right to demonstrate qualification by shielding, moving, determining that the components are not required for mitigation, demonstrating that the component has accepted failure modes or demonstrating minimal impact should failure occur.

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103

EQUIPMENT ENVIRONMENTAL QUALIFICATION	FRC TASK 02.198	REACTOR TYPE BWR	PLANT NAME Big Rock Point
	PROJECT 02G-C5257-01		UTILITY Consumers Power Co.
SEP PLANTS	DOCKET 50-155	NRC TAC 12377	DATE/ENGINEER

PAGE 1

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER																		
	1	2	3	4	5	6	7	8	9	10	11A	11B	11C	12A	12B	13	14	15	16
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				
RELATIONSHIP TO TEST SPECIMEN		X												X	X		L		
AGING DEGRADATION EVALUATED					X				X		X	X	X	X	X	L	L	L	
QUALIFIED LIFE ESTABLISHED	L	X			X	X	X	X	X		X	X	X	X	X	L	L	L	
PROGRAM TO IDENTIFY AGING	L	X			X				X		X	X	X	X	X	L	L	L	
QUAL. FOR STEAM EXPOSURE																			
PEAK TEMPERATURE ADEQUATE																			
PEAK PRESSURE ADEQUATE																			
TEST DURATION ADEQUATE					X				X	X	X								
REQUIRED PROFILE ENVELOPED																			
QUAL. FOR SUBMERGENCE					X	X		X											
QUAL. FOR CHEMICAL SPRAY						X													
QUAL. FOR RADIATION							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)																		
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.																			
IV-A. QUAL. TEST SCHEDULE																			
IV-B. QUAL. NOT ESTABLISHED		O						O		O	O	O	O	O	O				
V. EQUIP. NOT QUALIFIED			O	O	O		O		O										
VI. QUAL. IS DEFERRED						O													O
REPLACEMENT SCHEDULE					*				R		*	*	*				*		

REV 1

Franklin Research Center A Division of The Franklin Institute The Benjamin Franklin Parkway, Phila. Pa. 19103	FRC TASK 02.198	REACTOR TYPE BWR	PLANT NAME Big Rock Point	PAGE 2
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Consumers Power Co.	
SEP PLANTS	DOCKET 50-155	NRC TAC 12377	DATE/ENGINEER	

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER																		
	17	18A	18B	19A	19B	20	21	22	23	24A	24B	25	26	27	28	29	30	31	32A
GUIDELINE REQUIREMENTS.	(DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)																		
EVIDENCE OF QUALIFICATION	X			X	X	X	X			X	X	X	X			X	X	X	
RELATIONSHIP TO TEST SPECIMEN																		X	
AGING DEGRADATION EVALUATED	X					X		L		X	X	X							
QUALIFIED LIFE ESTABLISHED	X			X	X	X		L		X	X	X				X	X	X	
PROGRAM TO IDENTIFY AGING	X			X	X	X		L		X	X	X							
QUAL. FOR STEAM EXPOSURE						X													
PEAK TEMPERATURE ADEQUATE						X													
PEAK PRESSURE ADEQUATE																			
TEST DURATION ADEQUATE																			
REQUIRED PROFILE ENVELOPED																			
QUAL. FOR SUBMERGENCE																			
QUAL. FOR CHEMICAL SPRAY																			
QUAL. FOR RADIATION				X	X														
BETA RADIATION CONSIDERED																			
TEST SEQUENCE																			
TEST DURATION (1 HOUR + FUNCTION)						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											
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			O	O	O	
V. EQUIP. NOT QUALIFIED						O	O												
VI. QUAL. IS DEFERRED		O	O						O					O	O				O
REPLACEMENT SCHEDULE							*			R	R		R						

REV 1

Franklin Research Center A Division of The Franklin Institute The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK 02. 198	REACTOR TYPE BWR	PLANT NAME Big Rock Point	PAGE 3
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Consumers Power Co.	
SEP PLANTS	DOCKET 50-155	NRC TAC 12377	DATE/ENGINEER	

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER																		
	32B	33	34	35	36	37	38A	38B	39A	39B	40	41	42	43	44	45	46	47	48
GUIDELINE REQUIREMENTS. (DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)																			
EVIDENCE OF QUALIFICATION			X	X		X	X	X	X	X	X		X	X	X	X			X
RELATIONSHIP TO TEST SPECIMEN																			
AGING DEGRADATION EVALUATED			X			X					X	L							
QUALIFIED LIFE ESTABLISHED			X			X	X	X	X	X	X	L	X	X	X	X	L	L	
PROGRAM TO IDENTIFY AGING			X			X	X	X	X	X	X	L	X	X	X	X	L	L	
QUAL. FOR STEAM EXPOSURE							X	X	X	X	X								
PEAK TEMPERATURE ADEQUATE																			
PEAK PRESSURE ADEQUATE																			
TEST DURATION ADEQUATE																			
REQUIRED PROFILE ENVELOPED																			
QUAL. FOR SUBMERGENCE													X	X	X	X			
QUAL. FOR CHEMICAL SPRAY							X	X	X	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																			
III. EXEMPT FROM QUAL.					O														
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
VI. QUAL. IS DEFERRED	O	O																	
REPLACEMENT SCHEDULE			R																

REV 1

Franklin Research Center A Division of The Franklin Institute The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK 02. 198	REACTOR TYPE BWR	PLANT NAME Big Rock Point	PAGE 4
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Consumers Power Co.	
SEP PLANTS	DOCKET 50-155	NRC TAC 12377	DATE/ENGINEER	

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER												
	-9	50	51	52	53	55	56	57	58	59	60	61A	61B
GUIDELINE REQUIREMENTS.	(DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)												
EVIDENCE OF QUALIFICATION	X	X		X	X	X	X	X	X		X	X	X
RELATIONSHIP TO TEST SPECIMEN				X	X			X					
AGING DEGRADATION EVALUATED								X					
QUALIFIED LIFE ESTABLISHED								X					
PROGRAM TO IDENTIFY AGING								X					
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								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													
II-B. QUAL. PENDING MODIFICATION													
II-C. QUAL. < PLANT LIFE/FRC REVIEW													
III. EXEMPT FROM QUAL.										O			
IV-A. QUAL. TEST SCHEDULE													
IV-B. QUAL. NOT ESTABLISHED													
V. EQUIP. NOT QUALIFIED	O	O		O	O	O	O	O	O		O	O	O
VI. QUAL. IS DEFERRED			O										
REPLACEMENT SCHEDULE												R	R

REV 1

6. REFERENCES

1. C.F. Russo
Test Report: Sphere Elastical Penetration Assembly and 3-B/3855, April 17, 1958, Test Report of Sphere Electrical Penetration Assembly Units
Pittsburgh Testing Lab, 13-Mar-58
1-C/3855

2. Specification E-2345-104, Test of Typical Sphere Electrical Penetration Assembly Units
Bechtel Corp., 06-Feb-58
2345

3. W.C. Frederick and G.M. Rhodes
Design Qualification Report for Electrical Penetration Assemblies, Dresden Station Unit 1
Conax Corp., 25-Aug-80
IPS-389, Proprietary

4.1 R.B. Cherba
Document Transmittal to H.J. Palmer, D.P. Blanchard, and D.E. Moeggenberg, Subject: Containment Isolation Solenoid Valves, with memos dated 4/8/75 and 4/10/75
Consumers Power Co., 14-Apr-75

4.2 R.B. Sewell
Letter of Transmittal to NRC, with attached Special Report No. 21, Subject: Environmental Qualification Tests of Safety Related Equipment
Consumers Power Co., 15-May-75

4.3 R.B. Cherba and P.F. Johnson
Memo to C.J. Hartman, with 3 attachments. Subject: Radiation Resistance of Big Rock Control Cables
Consumers Power Co., 22-May-75
BRP-TF-34

4.4 J.D. Westbrook
Memo to R.B. Cherba, Subject: Junction Box Qualification
Consumers Power Co., 29-May-75
BRP-TF-222

4.5 M.R. Wade
Memo to W.S. Skibitsky, NRC, Subject: Environmental Qualification of Big Rock Point Original Containment Penetrations
Consumers Power Co., 16-Feb-78

4.6 Test Report: Size 22-16 PIDG Terminals
AMP Labs, 14-May-74
GPR-575-98

4.7 R.B. Cherba
Memo to C.J. Hartman, Subject: Control Cable Environmental Test Results and Recommendations, with attachments
Consumers Power Co., 22-May-75
BRP-TF-185, Proprietary

4.8 Target Rock Corp.
Environmental Test Report on Depressurization System Solenoid Valves
Target Rock Corp., 11-Aug-76
Report 1500, Proprietary

4.9 L.E. Witcher, W.H. Steigelmann and S.P. Carfagno
Report: Qualification Tests of Electrical Cables Under Simulated Post-Accident Reactor Containment Service Conditions, plus Addendums and letter of transmittal
FIRL, 15-Apr-70
Report No. F-C2737, Proprietary

4.10 N.M. Burstein & L.E. Witcher
Report: Qualification Tests of Electrical Cables Under Simulated Reactor Containment Service Conditions Including LOCA While Electrically Energized
FIRL, 01-Mar-74
Report No. F-C3798, Proprietary

4.11 Test Report: Subject: Tests of Raychem Flamtrol Insulated and Jacketed Electrical Cables under Simultaneous Exposure to Heat, Gamma Radiation, Steam, and Chemical Spray
FIRL, 01-Jan-75
Report No. F-C4033-1, Proprietary

4.12 L.E. Witcher, R.G. Yaeger, and D.V.Paulson
Test Report, Subject: Performance Qualification Tests of Four Valve Motor Operators
FIRL, 01-Apr-75
Report No. F-C4124, Proprietary

4.13 C.H. Heil
Test Report, Subject: Seismic Testing of ITT Barton Model 393 Static Pressure Electronic Transmitter
ITT, 28-Feb-72
P-1-386-6, Proprietary

4.14 L.D. Kenworthy
Test Report, Subject: Yarway Corp. Remote Level Indicator Switch
NUS Corp., 01-Mar-75
NUS-TM-ED-116, Proprietary

4.15 R.B. Cherba
Memo, Subject: Radiation Evaluation/Qualification of Limitorque and Rotorque Motor Operators
Consumers Power Co., 29-May-75
BRP-TF-220, Proprietary

4.16 W.S. Skibitsky
Licensee Event Reports, Subject: Emergency Diesel Generator Starting Time
Consumers Power Co., 01-Mar-79
RO-78-04,05,10

4.17 F.V. Thome
Report, Subject: Testing to Evaluate Synergistic Effects from LOCA Environments
Sandia Labs, 01-Apr-78
SAND78-0718, Proprietary

4.18 R.B. cherba
Memo to C.J. Hartman, Subject: Modification of Yarway Level Switch
Consumers Power Co., 21-May-75
BRP-TF-188

4.19 Internal Report, Subject: Big Rock Point Containment Isolation System
Consumers Power Co., 21-May-75
BRP-TF-172

4.20 A.J. DeGrasse
Letter to CPC, Subject: Qualification of Isolation Valve Solenoids
Catalytic, Inc., 17-Feb-78
C-36690-6

4.21 W.J. Beckius
Memo to W.S. Skibitsky, Subject: Electrical Equipment List for Environmental Qualification-SEP
Consumers Power Co., 08-Feb-78

4.22 A.O. Barry
Memo to R.E. Basso, Subject: Qualification Methods for Electrical Equipment
Catalytic, Inc., 22-Jun-76

4.23 G. Miller
Certificate of Compliance, Subject: Qualification of Pressure Switches
Pressure Switch Co., 02-May-75
7101-156

4.24 H.J. Palmer
Memo to R.W. Cherba, Subject: Radiation Qualification of Static-O-Ring Pressure Switches
Consumers Power Co., 23-May-75
BRP-TF-199

4.25 Letter to CPC, Subject: Temperature Qualification of Static-O-Ring Pressure Switches, with attachments
Pressure Switch Co., 14-May-75

4.26 R.B. Cherba
Memo to C.J. Hartman, Subject: Big Rock Point Junction Box Recommendations
Consumers Power Co., 30-May-75
BRP-TF-223

4.27 T.P. Schaefer
Test Report: Maximum Emergency Environmental Test of Electrical Penetration Assemblies
Conax Corp., 13-May-75
1PS-73.4, Proprietary

4.28 J. Lowe
Statement of Compliance with Specification I-1, for E10 Series Transmitter
Foxboro Co., 28-Apr-75
Proprietary

4.29 D.A. Savage
Certificate of Compliance for Differential Pressure Transmitter
Rosemount, Inc., 12-Nov-76
Proprietary

4.30 R.F. Powell
Certificate of Compliance for Digital Panel Meter Type PI 2455
Analogic Corp., 09-Feb-77

4.31 N.M. Burstein and L.E. Witcher
Report: Qualification Testing of Veritrak Pressure Transmitters for Service in a Nuclear Reactor Containment Facility
FIRL, 06-Oct-73
F-C3715, Proprietary

4.32 Product Data Sheet for 1151GP Alphaline Pressure Transmitter
Rosemount, 01-Jan-76
PDS 2260

4.33 A.O. Barry
Memo to R.E. Basso, Subject: Qualification Methods for Electrical Equipment
Catalytic, Inc., 22-Jun-76

4.34 D.A. Bixel
Letter to D.K. Davis, NRC, Subject: Environmental Qualification
Consumers Power Co., 24-Feb-78

4.35 R.A. English
Memo to D.G. Hauster, Subject: Audit of Dose Analysis for Big Rock Point Containment Following LOCA
Consumers Power Co., 20-May-75

4.36 R.W. Sinderman and W.R. Strodl
Memo to R.B. Cherba, Subject: Radiation Dose in Containment LOCA with Fuel Clad Perforation
Consumers Power Co., 10-Feb-75

4.37 Post-Incident Equipment Review, Subject: Breakers 15 and 16
Consumers Power Co., 29-May-75
BRP-TF-21.2

4.38 Post-Incident Equipment Review, Subject:
Flow Transmitter, Model 297, with drawings
Consumers Power Co., 23-May-75
BRP-TF-K2.6

4.39 Product Bulletin for Models 288A and 289A Indicating
Switches
ITT Barton, 01-Jan-77

4.40 W.C. Cooper
Memo to H.J. Palmer, Subect: Investigation of
Instruments in Big Rock Point Plant
Consumers Power Co., 17-Mar-75

4.41 Instruction Book: Multi-Point Indicator Gage
Bailey Meter Co., 14-Mar-62

4.42 Post-Incident Equipment Review, Subject: Switches
Consumers Power Co., 22-May-75
BRP-TF-K4.1

4.43 Post-Incident Equipment Review: SOLA Transformer
Consumers Power Co., 21-Apr-75

4.44 Post-Incident Equipment Review: Static-O-Ring
Pressure Switch
Consumers Power Co., 22-May-75
BRP-TF-K7.2

4.45 J.R. Yope
Memo to T.L. Randolph, Subject: Request for Quotation
for BRP Post Incident Cooling System, with Specification Sheet
Consumers Power Co., 05-Mar-75
YPE 75-1

4.46 D.E. Moeggenberg
Report: Environmental Conditions, Before, During, and
After LOCA
Consumers Power Co., 23-May-75
BRP-TF-A.30

4.47 P. Kelsa
Certificate of Compliance for Rosemount Power Supply
Rosemount Nashville, Inc., 19-Oct-76

4.48 Instruction Manual for Indicating Dual Alarm Instruments
International Instruments, 01-Oct-76
9270-20

4.49 A.O. Barry
Memo to R.E. Basso, Subject: Qualification Methods for Electrical Equipment
Catalytic, Inc., 22-Jun-76

4.50 W.S. Skibitsky
Letter to D.L. Ziemann, NRC, Subject: Environmental Qualification of Safety-Related Equipment, with attachments
Consumers Power Co., 23-Mar-78

4.51 D.E. Moeggenberg
Memo to D.A. Bixel, Subject: Big Rock Point EEQ Summary Sheet
Consumers Power Co., 05-Sep-78
MOEG 3-78

4.52 C.I. Hartman and D.E. DeMoor
Memo to R. Marusich, Subject: Update of SEP Tables for Equipment Qualification
Consumers Power Co., 02-Oct-78
DED 78-37

4.53 R. W. Sinderman
Memo to F.W. Buckman, Subject: Dose Assessment for POST-LOCA Direct Shine
Consumers Power Co., 25-Jan-78
RWB 11-78

4.54 R.B. Cherba
Memo to C.J. Hartman, Subject: Qualification of Pressure Transmitter Loop PT 186
Consumers Power Co., 21-May-75

4.55 R.G. Schantz
Test Report: Elevated Temperature Test of AMP "Certi-Seal" Wire Connectors
Consumers Power Co., 02-Apr-75
I-8-A-75, Proprietary

4.56 Purchase Order, Spec Sheet, and Certificate of Compliance for Reactor Depressurizing System Terminal Boxes
Catalytic, Inc., 22-Jul-75
34490-1601-50

4.57 Report No. I: Effect of Compartment Pressurization due to Pipe System Break Outside Containment
Consumers Power Co., 01-Apr-73

4.58 Report No. II: Effect on Safety Related Equipment Due to Pipe System Break Outside Containment
Consumers Power Co., 25-Apr-73

4.59 R. Marusich
Telephone log: Call to B. Eisler, Subject: Radiation Resistance of Electrical Insulation
Consumers Power Co., 19-Oct-78

4.60 D.A. Bixel
Letter to D.K. Davis, NRC, Subject: Fire Safety Analysis and Proposed Technical Specifications Change Request, with attached reports
Consumers Power Co., 29-Mar-77

4.61 R.M. Marusich
Memo, Subject: Qualification of the Cable Splices which were Covered with Waterproofing Tape and Epoxy Resin Potted Splices
Consumers Power Co., 30-Oct-78
MARV 3-78

4.62 M.J. Aklinski
Test Report: Static-O-Ring Pressure Switches
M.E.A. Inc., 10-Nov-72

4.63 W.M. Brown
Letter to Catalytic, Inc., Subject: Qualification of Valve at High Ambient Temperatures
Automatic Switch Co., 11-Jun-76
Proprietary

4.64 C.J. Hartman
Memo: Comments on SEP Draft
Consumers Power Co., 03-Nov-78
DED 78-40

4.65 Bechtel Corp.
Report: Design, Fabrication, Testing, and Test Results of Typical Sphere Penetrations
Bechtel Corp., 07-Jul-58
Proprietary

4.66 M.S. Turbak
Letter to E.G. Case, NRC, Subject: Testing of Electrical Penetration, with drawings
Commonwealth Edison, 09-Dec-77

4.67 R.O. Aumiller
Letter to Bechtel Corp., Subject: Testing of Electrical Penetrations
Amphenol, 21-Sep-61

4.68 B.H. Randolph
Letter of Transmittal: Electrical Penetration Plates, with drawings
Amphenol, 01-Mar-62

4.69 R.M. Marusich
Memo: Radiation Qualification of Amphenol Borg Penetration Units
Consumers Power Co., 29-Nov-78
MARV 3-78

5. W. Bernbeck, G.L. Meiler, and W.S. Raulio
Test Report: Electrical Terminations Subjected to Design Basis Accident Environment, North Anna Power Stations I and II
Conax Corp., 05-Oct-73
IPS-107

6.1 T. Hess, W. Denkowski, and C. Formica
Appendix D of Outside Containment Qualification Report
Limitorque Corp., 00-Jun-76
Report No. B0003

6.2 T. Hess, W.J. Denkowski, and C.D. Formica
Appendix E of DC Actuator Qualification Report
Limitorque Corp., 30-Apr-76
Report No. B-0009

7. Report: Nuclear Qualification of Valve Actuators
Limitorque Corp., 27-Aug-80
Report No. B0058

8. J.S. Lasky
Letter: C.A. St. Orge (Bechtel), Subject: Environmental Qualification of Cables with Report NQRN-1
Ononite Co., 23-Sep-80

9. D.A. Shiells, D.W. Halligan, and B.H. Randolph
Report: Final Integrity Test and Leakage Rate Determination, Reactor Containment Vessel, Big Rock Point Nuclear Plant
Bechtel Corp., 00-Jul-62
3159

10. C.J. Hartman (CPC)
Deviation Report: Traceability for Test Switch, Yarway, with attachments
Consumers Power Co., 05-Oct-78
DAD-100678-2

11. R.M. Marusich (CPC)
Memo: Aging of Neoprene and Asbestos, with Wyle Labs Report No. 17435-1, dated August 4, 1980
Consumers Power Co., 15-Oct-80

12. W.J. Denkowski and C.D. Formica
Qualification Type Test Report: Limitorque Valve Actuators for BWR Service
Limitorque Corp., 13-May-76
Report No. 600376A

13. L.D. Kenworthy and G.C. Rudy
Test Report: Yarway Corp. Remote Level Indicator Switch Model 4416C
NUS Corp., 00-Mar-75
Report No. NUS-TM-ED-116

14. R.E. Trautwein and F.W. Bach
Test Report: Yarway Corp. Remote Level Indicator Figure Nos. 4418TC S/N 35364
Lockheed Electronics Co., 27-Mar-79
Report No. 5628-3509

15. R.F. Soltis and F.W. Bach
Test Report: Yarway Corp. Remote Level Indicator
Figure Nos. 4418EC
Lockheed Electronics Co., 22-Mar-73
Report No. 3232-3155

16. G. Tugwell
Heat Aging Class B NA2 Motor For Outside Containment
Safety Related Duty
Rotork, 21-Oct-77
Report No. TR422

17. B. Gregory and G. Crook
Report: Actuation Radiation Subjection
Rotork, 00-May-70
Report No. N14/2

18. Post-Incident Equipment Review, Big Rock Point
Consumers Power Co.
K5.1

19. H.J. Palmer (CPC)
Memo to R.B. Cherb, Subject: Jo-Bell Level Switches
Qualification
Consumers Power Co., 20-May-75
BRP TF 182

20. H.J. Palmer
Memo to R.B. Cherba (CPC), Subject: Static "O" Ring
Pressure Switches Radiation Qualification
Consumers Power Co., 23-May-75
BRP TF 199

21. L. Kinne, A.W. Libbe, J.A. Jail, and
J.J. Juergens
Qualification Tests: Rosemount Pressure Transmitter
Model 1152, Rev. B
Rosemount Inc., 26-Oct-76
Report No. 117415

22. J. Fitzsimmons
Engineering Report: Valves (LWO's 7765 and L7947)
Consumers Power Co., 27-Feb-73
Report No. 47

23. Test Report: TRC Model 72V Type, 1" Y-Pattern Solenoid Motor Valve per Provisions of TR Test Procedure 1383
Target Rock Corp., 22-Oct-74
Report No. 1500

24. R.H. Dille, K.V. Allen, H.W. Pielage, and A.W. McCrae
Report: Cable Splice Waterproofing Environmental Test, Fire Protection Modification for Big Rock Point Plant
Nuclear Services Corp., 08-Sep-78
Report No. CPC-0116 609153

25. W.R. Lankenau
Test Report: The Effect of a LOCA on the Electrical Performance of Four Terminal Blocks
Westinghouse, 13-Sep-77
Report No. PEK-TR-77-83

26. D.B. Vail
Memo to File Subject: Terminal Block,Junction Box Environmental Testing Report
Northeast Utilities, 27-Mar-78
GEE-78-127

27. Engineering Test Report: AMP RAdiation Resistant PIDG Terminals
AMP Incorporated, 14-May-74
GPR 575-98

28. N.M. Burstein & L.E. Witcher
Report: Qualification Tests of Electrical Cables Under Simulated Reactor Containment Service Conditions Including LOCA While Electrically Energized
FIRL, 01-Mar-74
Report No. F-C3798, Proprietary

29. Report: Tests of Electrical Cables Subjected to Thermal Aging, Gamma RAdiation and a Loss-Of-Coolant Accident Simulation
FIRL, 00-Jul-76
Report No. F-C4350-3

30. Report: Qualification Tests of Electrical Cables in a Simulated Loss-of-Coolant-Accident (LOCA) Environment
FIRL, 00-Apr-79
F-C5115

31. L.E. Witcher, W.H. Steigelmann and S.P. Carfagno
Report: Qualification Tests of Electrical Cables Under Simulated Post-Accident Reactor Containment Service Conditions, plus Addendums and letter of transmittal
FIRL, 15-Apr-70
Report No. F-C2737, Proprietary

32. Performance Qualification Tests of Four Valve Motor Operators
FIRL, 00-Apr-75
Report No. F-C4124

33. J.P. Waggener and L.E. Witcher
Performance Test of three Differential Pressure Transmitters in a Simulated Reactor Containment Post Accident Steam Environment
FIRL, 00-Nov-69
Report No. F-C2667, App. B, Proprietary

34. N.M. Burstein and L.E. Witcher
Report: Qualification Testing of Veritrak Pressure Transmitters for Service in a Nuclear Reactor Containment Facility
FIRL, 06-Oct-73
F-C3715, Proprietary

35. Test Report: Subject: Tests of Raychem Flamtrol Insulated and Jacketed Electrical Cables under Simultaneous Exposure to Heat, Gamma Radiation, Steam, and Chemical Spray
FIRL, 01-Jan-75
Report No. F-C4033-1, Proprietary

36. J.F. Kirchner and R.E. Bowman
EFFECTS OF RADIATION ON MATERIALS AND COMPONENTS, pp 118-129, and pp 369-371
Reinhold Publishing Corp.

37. L. Clarke and R.L. DAvidson
MANUAL FOR PROCESS ENGINEERING CALCULATIONS, 2nd edition, pp. 70-75
McGraw-Hill Book Co.

38. Product Bulletin: Differential Pressure Electronic Transmitter Models 332, 333, 367, 368, 384, 385, 396 and 397
ITT Barton, 01-Jan-77
332-2

39. J.D. Gratz
Report: LOCA Qualification of ACRO Switch, Big Rock Point Nuclear Plant, M&S Test Procedure 51, Rev. A
Consumers Power Co., 10-Oct-78
Report No. 378018-10A

40. A.P. Colaiaco
Material Evaluation: Westinghouse Terminal Blocks
Westinghouse

41. Nuclear Differential Pressure Transmitter - Model 59DP Series, Instruction Bulletin
Westinghouse Electric, 00-Jun-74
IB-101-191

42. Descriptive Bulletin: Terminal Blocks Application
Westinghouse
34-350

44. Model Number System for Housings, Switch Elements, Springs, and Options
Static "O" Ring, 00-Apr-72

45. G. Mills
Certificate of Compliance for Pressure Switch
Static "O" Ring, 12-May-75
7101-156

46. Product Data Sheet: Model 1151P Alphaline Gauge Pressure Transmitters
Rosemount Inc., 1976
2260

47. D.A. Elerage
Quality Certification of Compliance Data Sheet: Differential Pressure Transmitter 1152
Rosemount Inc., 12-Nov-76

48. Specification Sheet: High Range Survey Meter, Underwater
Model CP-MU
Technical Associates

49. R.W. King et al.
Report: The Effects of Nuclear Radiation on Elastomeric and Plastic Components and Materials
REIC
Report No. 21

50. D.A. Bixel (CPC)
Letter: J.G. Keppler (NRC), Subject: Deterioration of Buna-N Components in ASCO Solenoids
Consumers Power Co., 02-Feb-79

51. G.G. Littlehales
Certification of Conformance for Suntac Nuclear Corp.
Cerro Wire & Cable, 13-Jun-75
026/2205

52. Qualification of Firewall III Class 1E Electric Cables (Chemically Cross-Linked Insulation)
Rockbestos Co., 26-Nov-79

55. R.B. Bladgett and R.G. Fisher
Insulations and Jackets for Control and Power Cables in Thermal Readtor Nuclear Generating Stations,
Vol. PAS-88, No. 5
IEEE Transactions, 00-May-69
68TP651-PWR

57. J.C. Carroll and J.R. Maher
Continued Evaluation of Butyl - Rubber-Insulated Cable, (after 1959), pp. 10-23
AIEE Transactions

58. N.D. Kenney, T.N. Mitropoulos, and W.L. Seamonds
Electrical Characteristics of Butyl Insulation at 125 C, (after 1959), pp. 3-9
AIEE: Transactions

59. L.W. Ricketts
FUNDAMENTALS OF NUCLEAR HARDENING OF ELECTRONIC EQUIPMENT, pp. 118-122
Wiley-Interscience

60. W.S. Rantio
Letter to C.A. St. Onge (Bechtel), Subject: Conax Electric Penetration Assemblies, Big Rock Nuclear Power Station
Conax Corp., 30-Oct-80

62. R.M. Marusich (CPC)
Letter to J. Doyon (ITT Barton), Subject: Models 332 and 386 Pressure Transmitters
Consumers Power Co., 07-Nov-80

64. R.M. Marusich (CPC)
Letter to W. Maurer (Jo-Bell Products), Subject: Telephone conversation of Oct. 14, 1980
Consumers Power Co., 16-Oct-80

66. D.P. Hoffman
Letter: J.G. Keppler (NRC), with LER-80-023 and Reanalysis for Steam Phase Breaks
Consumers Power Co., 26-Aug-80

67. R.B. Cherba (CPC)
Memo to C.J. Hartman (CPC), Subject: Yarway Level Switch Modification LS-RE09A-H
Consumers Power Co., 20-May-75
BRP TF-188

68. R.B. Cherba and H.J. Palmer
Memo C.J. Hartman (CPC), Subject: Yarway Level Switch Tests and Modifications
Consumers Power Co., 19-May-75
BRP TF 181

69. R.H. Arnold
Letter to J. Kuemin (CPC), Subject: Qualification Data on Sealed Valve Actuators, with attachments
Rotork Controls, 17-Oct-80
A538

71. M.J. Aklinski
Letter: D.L. Giuliani (Conroy Mechanical Contractors), with Static "O" Ring Test Report 7201-105 for pressure switches
M.E.A., Inc., 10-Nov-72
1495

72. R.M. Bowman
Letter: J.D. Westbrook (CPC), Subject: Cable Information
Kerite Co., 17-Mar-75

73. D.J. Blaies
Letter: G. Blickley (Mechni Arts Associates), Subject: Switch MT-4R
Micro Switch, 16-May-75

74. H.E. McCrane
Letter: Consumers Power Co., Subject: FR Insulation and Jacket Materials
Kerite Co., 29-May-75

75. M.A. Costandi
Letter: J. Yope (CPC), Subject: Electrical Terminal Block Testing
General Electric, 21-Feb-79
G-EJ-9-17

76. W.S. Skibitsky
Letter: J.G. Keppler (NRC), Subject: Response to IE Bulletin 78-02
Consumers Power Co., 13-Feb-78

77. H. Kenny
Letter: E.R. Longman (CPC), Subject: Order 32277-Q Data, with Attachments
Anaconda Co., 26-Dec-78

78. R.O. Bolt and J.G. Carroll
RADIATION EFFECTS ON ORGANIC MATERIALS, pp 456-459 (1963)
Academic Press, NY, 01-Jan-63

79. HANDBOOK OF CHEMISTRY AND PHYSICS, 53rd Edition, p. F-134
CRC Press, 01-Jan-73

80. C.A. St. Onge
Letter to W. Beckins (CPC), Subject: Properties of Materials Used for Electrical Penetrations
Bechtel Associates, 30-Oct-80
0275

81. J.F. Kircher and R.E. Bowman
EFFECTS OF RADIATION ON MATERIALS AND COMPONENTS, pp 96-97
Reinhold Publishing Corp.

82. J.R. More
Letter: W. Hall (CPC), Subject: Radiation Test Results
3M Company, 20-Oct-78

83. L. Clarke and R.L. Davidson
MANUAL FOR PROCESS ENGINEERING CALCULATIONS, 2nd Edition, p. 73
McGraw-Hill Book Co.

84. G.Y. Chinn, W.C. Frederick, F.J. Illig, and G.M. Rhodes
Design Qualification Material Test Report for Conax Electric Penetration Assemblies and Electric Conductor Seal Asemblies
Conax Corp., 02-Nov-78
IPS-325, Proprietary

85. G. Lainas (NRC)
Memo to A. Schwencer (NRC), Subject: Electrical Equipment Environmental Qualification, with 2 attachments containing DOR Guidelines.
USNRC, 19-Feb-80

86. FRC Draft Interim Technical Evaluation Report on Equipment Environmental Qualification for Big Rock Point Plant.
Franklin Research Center, 23-Oct-80

87. N.C. Moseley (NRC)
Letter to B.H. Grier (NRC), Subject: IE Supplement No. 2 to Bulletin 79-01B, Environmental Qualification of Class 1E Equipment
NRC, 29-Sep-80

88. N.C. Moseley (NRC)
Letter to B.H. Grier (NRC), et al., Subject: Supplement No. 3 to Bulletin 29-01B, Environmental Qualification of Class 1E Equipment
USNRC, 24-Oct-80

89. S.J. Chilk (NRC)
Memorandum and Order pursuant to Union of Concerned Scientists Petition for Emergency and Remedial Relief
USNRC, 23-May-80
CLI-80-21

90. D.P. Hoffman
Letter to D.M. Crutchfield (NRC) with attached Report, Environmental Qualification of Safety-Related Electrical Equipment, Big Rock Point Plant
Consumers Power Co., 21-Oct-80

91. R.B. Sewell
Letter to Division of Reactor Licensing, NRC.
Consumers Power Co., 15-May-75

92.1 Letter to C.J. Crane (FRC), Subject: System and Component Listing
Consumers Power Co., 22-Sep-80

92.2 D.E. Moeggenberg (CPC)
Letter to C.J. Crane (FRC), Subject: Request for Deferral of Post-Incident Sampling System, with attachment
Consumers Power Co., 15-Sep-80

92.3 Letter to D.L. Ziemann (NRC), Subject: Revisions to EEQ tables of Nov. 30, 1980
Consumers Power Co., 20-Dec-78

92.4 D.A. Bixel (CPC)
Letter to D.L. Ziemann (NRC), Subject: Submittal of EEQ Information, with Attachment
Consumers Power Co., 30-Nov-78

92.5 Letter to D.K. Davis (NRC), Subject: Big Rock Point Environmental Qualification, with attachment
Consumers Power Co., 24-Feb-78

92.6 D.M. Crutchfield (NRC)
Memo to Z.R. Rosztoczy (NRC), Subject: Big Rock Point Site Visit Environmental Qualification Temperature Profile
USNRC, 16-Sep-80

92.7 Drawings, Big Rock Point Nuclear Plant, Information Copy No. 1
Consumers Power Co., 29-Feb-80

92.8 V. Stello (NRC)
Letter to D.A. Bixel (CPC), subject: Initiation of EEQ Review
USNRC, 23-Dec-77

92.9 W.S. Skibitsky (CPC)
Letter to D.L. Ziemann (NRC), Subject: Corrections to EEQ Evaluation of Feb. 24, 1978, with attachment
Consumers Power Co., 23-Mar-78

93.10 Teleconn to Westinghouse WCID, Subject: Review of Westinghouse Model Numbers for Component Similarity
FRC, 15-Oct-80

93.11 Telecon to Consumers Power Co., Subject: Big Rock Point Plant Questions and Clarifications
FRC, 15-Oct-80

93.12 D.E. Moeggenberg (CPC)
Letter to C.J. Crane (FRC), Subject: Additional Environmental Parameters
Consumers Power Co., 30-Sep-80

93.13 D.E. Moeggenberg (CPC)
Letter to J. Archer (FRC), Subject: Additional Environmental Parameters
Consumers Power Co., 20-Oct-80

93.14 D.E. Moeggenberg (CPC)
Letter to J. Archer (FRC), Subject: Clarification of Submitted Documentation
Consumers Power Co., 17-Oct-80

93.15 Letter to J. Archer (FRC), Subject: Transmittal of Test Reports, Lockheed Electronics, Nos. 5628-3509, and 3232-3155
Yarway Corp., 16-Oct-80

94. D.E. Moeggenberg
Letter to C.J. Crane, Subject: References in 10/31/80 Submittal to NRC: Electrical Equipment Qualification for Big Rock Point Plant
Consumers Power Co., 12-Nov-80

95. D.M. Crutchfield
Letter to D.P. Hoffman (CPC), Subject: Environmental Qualification of Electrical Equipment (Big Rock Point and Palisades Plant)
USNRC, 29-Aug-80

96. D.L. Ziemann
Letter to D.P. Hoffman (CPC), Subject: Electrical Equipment Environmental Qualification, Palisades/Big Rock Point Plants, with attachments
USNRC, 15-Feb-80

97. D.L. Ziemann (NRC)
Letter to D. Bixel (CPC), Subject: Control Rod Failure
USNRC, 06-Sep-78

98. K.R. Goller (NRC)
Letter to R.B. Sewall (CPC), Subject: Approval of Operation of Big Rock Point in Accordance with Operating License No. DPR-6
USNRC, 30-May-75

99. R.W. Sinderman (CPC)
Memo to D.E. Moeggenberg (CPC), Subject: Review of Radiation Doses, with attachments
Consumers Power Co., 20-Oct-80
RWS 80-122

100. K.R. Galler
Letter to R.B. Sewall (CPC), Subject: Amendment No. 8 to Facility License No. DPR-6, and Safety Evaluation
USAEC, 05-Nov-74

101. S.A. Giusti
Letter to T.A. Bjornard (CDC), Subject: Pipe Break Evaluation - Compartment Pressurization, File 0270 & 2200
Bechtel Power Co., 08-Mar-73

102. G.J. Walke
Letter to J.F. O'Leary (AEC), Subject: Change No. 39 to Technical Specifications, with attachments
Consumers Power Co., 29-Jun-73

103. R.B. Sewall
Letter to USNRC, Subject: Additional Responses to Special Report No. 21
Consumers Power Co., 15-May-75

104. D.A. Bixel
Letter to J.G. Keppler (NRC), Subject: Response to IE Report 050-155/77-04
Consumers Power Co., 11-May-77

105. Loss of Reactor Coolant, EMP 3.3
Consumers Power Co., 15-May-80
Rev. 12

106. Post-Incident System, System Operating Procedures, SOP-8
Consumers Power Co., 19-May-80
Rev. 13

107. Reactor Building Temperatures, Jan. through Aug. 1978
Consumers Power Co., 00-Jan-78

108. Telecon between FRC and NRC, Subject: Big Rock Point Plant Environmental Parameters for Containment. 21-Jan-81

109. S.R. Frost (CPC)
Letter to J.G. Keppler (NRC), Subject: Licensee Event Report 80-048 -- Electrical Equipment Qualification.
Consumers Power Co., 30-Dec-80

110. W. N. Lawford and D.L. Ham
Radiation and Seismic Testing of Barton Model 386 Pressure Transmitter
ITT Barton, 17-May-71
Report No. R1-386-2

111. E.A. Lommatsch
Letter to R. Marusich (CPC), Subject: FIRL Report F-C2667
ITT Barton, 28-Oct-80

112. R.M. Marusich (CPC)
Memo: R.E. Schrader (CPC), Subject: Sealing of Containment Level Switches LS-3562 through -3565
Consumers Power Co., 16-Oct-80
MARU 70-80

113. B. May (CPC)
Memo: GE Motor Nameplate Data
Consumers Power Co., 25-Oct-80

114. J.A. Olshinsk (NRC)
Letter to D.M. Crutchfield (NRC), Subject: Big Rock Point - Modifications to Containment Spray System (TAC-42812) Safety Evaluation Report.
USNRC, 29-Dec-80

115. R.M. Marusich (CPC)
Letter: D. Buling (GE), Subject: Core Spray Pump Motor Nameplate Data
Consumers Power Co., 27-Oct-80

Appendix A - ENVIRONMENTAL SERVICE CONDITIONS

The Licensee has provided information in Reference 90 concerning environmental service conditions in various locations within the plant. The containment temperature and pressure response (Figure A-1) following the postulated loss-of-coolant accident (LOCA) was provided by the Licensee, [90] based on the most recent reanalysis of post-accident containment pressure. [90]

For BWR plants, the DOR Guidelines state that the environmental service conditions inside the drywell for the LOCA should be considered to be 340°F for 6 hours.

However, Supplement 2 to IE Bulletin 79-01B provides the following additional criteria: "For minimum high temperature conditions in pressure-suppression-type containments, it is not required that 340°F for 6 hours be used for BWR drywells or that 340°F for 3 hours be used for PWR ice condenser lower compartments. These values are a screening device 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."

This criterion allows the Licensee to submit plant-specific profiles which account for the LOCA and high energy line breaks (HELB) inside the primary containment.

FRC notes that the design of the Big Rock Point reactor does not incorporate a pressure-suppression type containment. Consumers Power Company is currently in the process of calculating the plant-specific containment temperature/pressure profile resulting from a complete spectrum of potential line breaks. Preliminary results for these calculations indicate that a main steam line break (MSLB) could cause excessive temperatures. Because of this preliminary analysis, the Licensee is investigating various methods of short-term resolution including revision to the plant emergency operating procedures and plant modifications. In addition, special humidity sensors and temperature detectors have been installed inside the containment as part of the short-term program for resolution.

It was decided at the September 8, 1980 site meeting with the NRC and Licensee that the LOCA pressure/temperature curve provided by Reference 5 would be used for the purpose of the equipment environmental qualification evaluation as an interim measure until the final profile is submitted to the NRC by the Licensee.

For plants equipped with automatic containment spray systems not subject to single component failure or delayed initiation, the Guidelines state that equipment qualified for the LOCA environment is also considered qualified for the postulated MSLB accident. The design of the Big Rock Point Reactor does not satisfy these criteria. The containment spray system consists of two spray trains; one train is automatically started after a 15-minute time delay, the other train is locked out of operation. In addition, the spray system is subject to a single passive failure.

The NRC has determined [108] that the most recent Licensee analysis of environmental service conditions is to be used by FRC for this equipment environmental qualification review. Based on these considerations, FRC has evaluated each equipment item with respect to the environmental service conditions presented in this appendix. Environmental parameter values not referenced are assumed values.

The specific environmental service conditions corresponding to different plant locations that were used in this technical evaluation are stated in this appendix. The environmental zones within the plant are shown in drawing M-100 and Figures A-2 and A-3.

A.1 Inside Containment [90]

Normal Operation [Assumed Values]:

Temperature	50-90°F
Pressure	Atmospheric (14.7 psia)
Humidity	20-80%

Accident Conditions [90]

Temperature	See Figure A-1
Pressure	See Figure A-1
Chemical	None
Humidity	100%
Submergence	Elevation 590'

Radiation - Rads - 30 Days	2 Hours	24 Hours	30 Days
Gamma - Air	2.0×10^5	4.9×10^5	7.3×10^5
Gamma - Water	1.0×10^6	1.1×10^6	2.8×10^6
Beta - Air	1.3×10^6	4.2×10^6	1.3×10^7
Beta - Water	4.0×10^5	7.3×10^5	1.2×10^6

The Licensee stated: [90]

> The containment is subjected to pressure, temperature, humidity, gamma and beta radiation, submergence, containment spray, and some thermal aging during the course of an MSLB or a LOCA inside.

RADIATION INSIDE CONTAINMENT

The source terms used to calculate containment doses and all subsequent doses came from the ORIGEN code. The gamma and beta energies per disintegration for each nuclide are used. The beta energies are from the ORNL MEDLIST and the gamma energies were compiled from a listing for the computer program RASTUS. The RASTUS code calculates doses using Equation II as found in a letter from D. A. Bixel to J. G. Keppler dated May 11, 1977 regarding Consumers Power Company's response to IE Report 050-155/77-04.

Containment

Containment beta and gamma dose rates for air and liquid were calculated at t=0, 2 hours, 24 hours, and 720 hours. Infinite cloud calculations were performed to obtain dose rates in all cases except the gamma dose rate in air.

The 30-day integrated radiation doses were obtained by segmenting time periods and summing the products of each time period multiplied by the average logarithmic dose rate in that time period. The results are as follows:

Containment Air Beta	1.32E + 7 rads
Containment Air Gamma	7.35E + 5 rads
Containment Water Beta	1.15E + 6 rads
Containment Water Gamma	2.83E + 6 rads

Beta doses were applied to cable only as the equipment provides adequate shielding for beta radiation.

A.2 Pipe Tunnel: [1]

Normal Operation

Temperature	90°F-100°F
Pressure	Atmospheric
Humidity	90%
Radiation	Normal (outside containment)

Accident Conditions: [90]

	HELB	LOCA
Pressure	2.2 psig*	0 psig
Temperature	210°F*	90°F-100°F
Chemical	None	None
Humidity	100%	90%
Submergence	None	None
Radiation - Rads - 30 Days		
Gamma	4.9×10^3	Sphere Surface 7.6×10^4 6' From Sphere 5.2×10^4
Beta	7.0×10^3	None

The Licensee stated: [90]

> The Pipe Tunnel is located in the Turbine Building. It is an area where the fluid piping enters the Containment Building including the main steam and feedwater lines. Although some gamma radiation will be accrued by the equipment in the tunnel subsequent to a LOCA due to shine from the Containment wall, the MSLB provides a more severe environment to equipment in this area.

Beta and gamma 30-day integrated doses were calculated for the turbine building given the following initial assumptions:

a. The entire primary coolant volume (3,830 ft^3 from Technical Specification 4.1.2) is dumped to the turbine building and it all flashes to steam.

b. The main steam isolation valve closes before any fuel failure occurs and no fuel failure occurs after closure.

A.3 Electrical Penetration Room

Normal Operation [1]

Temperature	40°-90°F
Pressure	Atmospheric
Humidity	100%
Radiation	Normal (outside containment)

Accident Conditions [90]	HELB	LOCA
Pressure	0 psig	0 psig
Temperature	40°-100°F	182°F*
Chemical	None	None
Humidity	20%-100%	20% - 100%
Submergence	None	None
Radiation - Rads - 30 Days		
Gamma Negligible	Sphere Surface 7.6×10^4 rad	
	6' From Sphere 5.2×10^4 rad	
Beta Negligible	None	

*This temperature is taken from data in Computer code Safe C/BWR1 dated 6/6/75, run 5 S228 Case 2, 0.6303 ft^2 break. For time of duration, the temperature envelope in Figure 3 conservatively is used (i.e., draw a horizontal line on Figure 3 at 182°F).

*According to Computer code Safe C/BWR1 dated 6/6/75, run 5 S228 Case 2, 0.6303 ft^2 break, the low reactor water level scram (containment isolation signal) occurs after 46 seconds. The main steam isolation valve requires 60 seconds to close after the low reactor water level scram set point. According to letter from G. J. Walke, Consumers Powers, to J. F. O'Leary, NRC, dated June 29, 1973, the pressure peaks at 2.2 psig in 1.5 seconds. To summarize, the peak temperature and pressure occur in 1.5 seconds and are maintained for approximately 2 minutes before the steam source is isolated.

The Licensee stated: [90]

The Electrical Penetration Room is subjected to humidity during an MSLB outside containment. During a LOCA inside containment, the Electrical Penetration Room equipment receives a radiation dose due to shine from the containment wall. This room is unlike any of the other hostile areas outside containment in that it is insulated and, therefore, suffers a temperature rise through the containment wall.

The gamma dose rate due to containment shine in the electrical penetration room has been previously calculated. The methodology used was the same as referenced above (D. A. Bixel to J. G. Keppler, May 11, 1977). As was shown in that submittal, the electrical penetration room receives radiation from only 35% of the containment, as a result of its location relative to the internal structures. Credit is also taken for shielding from the containment shell. The 30-day integrated dose was obtained by the same method described under containment doses. The shine dose at six feet from the containment surface is 5.21E + 4 rads. The dose rates at six feet and on the containment surface were ratioed and multiplied by the 30-day integrated dose at six feet to obtain the 30-day integrated dose at the containment surface. The dose at the surface was found to be 7.6E+4 rads.

A.4 Sphere Ventilating Room

Normal Operation [1]

Temperature	60°-90°F
Pressure	Atmospheric
Humdity	60-90%
Radiation	Normal

Accident Conditions: [90]	HELB	LOCA
Pressure	0 psig	0 psig
Temperature	40°-90°F	40°-90°F
Chemical	None	None
Humidity	20%-100%	20%-100%
Submergence	None	None
Radiation - Rads - 30 Days		
Gamma	Negligible	Containment Surface 1.9×10^5
Beta	None	None

The Licensee stated [90]:

The Sphere Ventilating Room receives a gamma radiation dose due to shine from the containment wall during a LOCA. The contruction of the building is such that heat can be readily dissipated; therefore, an ambient temperature is assumed. The structure is physically removed

from the Turbine Building and will not be affected by an MSLB outside containment.

The radiation dose due to gamma shine at the air shed was calculated by using the following three volumes of activity. One is in the vicinity of the cooling unit, another in front of the clean-up demineralizer, and a third at the top half of the containment sphere. Using the whole top half of the containment sphere is conservative since the shielding around the steam drum would diminish the dose rate. The computer program RASTUS was used to calculate the dose rates from three equivalent volume spheres. The two smaller spheres were taken to be of equal volumes and distances to the air shed. The radius of the smaller spheres is 3.81E+2 cm with a distance of 3.45E+2 cm to the air shed. The larger sphere has a radius of 1.47E+3 cm with a distance of 2.07E+3 cm to the air shed. Shielding used for all three shields is 3/4" of iron.

The dose rates from the three spheres as provided by RASTUS (runs 478 through 485) were added for each time period. The integrated dose is 1.94E+5 rads.

A-5 Core Spray Room

Normal Operation: [1]

Temperature	60°-90°F
Pressure	Atmospheric
Humidity	60-90%
Radiation	Normal

Accident Conditions: [90]	HELB	LOCA
Pressure	0 psig	0 psig
Temperature	60°-90°F	152°F*
Chemical	None	None
Humidity	20%-80%	20%-80%
Submergence	None	None
Radiation - Rads - 30 Days		
Gamma Negligible		4.0×10^4 rad
Beta None		None

*This is the peak temperature reached in the core spray room 400 hours after the start of the recirculation mode. This temperature is constant for the remainder of the 30 days.

The Licensee stated: [90]

The Core Spray Room is a subterranean structure which houses the core spray pumps and core spray heat exchanger. The structure is physically removed from the containment wall and will, therefore, not be subjected to shine through the containment wall or a temperature rise from heat transfer through the containment wall; however, when the containment sump water is recirculated through the heat exchanger subsequent to a LOCA, the recirculated fluid provides a gamma radiation source and a heat source. An MSLB outside containment has no effect on this room.

The gamma radiation dose due to recirculating fluids in the Core Spray Room was calculated. The calculation involved modeling the piping carrying radioactive sources in the room by using spheres of diameters slightly larger than the pipe diameter to account for all of the volume. The dose in the room is then a sum of the contributions from all of these sources. This calculation gives a 30-day integrated dose of 2.64E+3 rads; however, it assumed a 10% core melt and no noble gases or particulates in the liquid phase. Regulatory Guide 1.3 calls for 100% of the noble gases and 1% of the particulates in the liquid phase.

To update this dose to the Regulatory Guide 1.3 source terms, it was first multiplied by ten for the 100% fuel melt case to give 2.64E+4 rads. To add the contribution from noble gases and particulates, a recent RASTUS run at t=2 hours for containment air was used to compare the contribution from noble gases and particulates vs halogens. The total dose rate due to noble gases and particulates was found to be roughly half of the total dose rate due to halogens (halogen dose rates were multiplied by 50 since the RASTUS run was for 1% of the halogens at t=2 hours and the liquid phase contains 50% of the halogens).

The total 30-day integrated dose is therefore:

$$(2.64E+4) + (2.64E+4)(0.5) = 3.96E+4 \text{ rads}$$

This is conservative since the dose rate ratio of noble gas and particulates to halogens becomes less at times greater than two hours.

Doses from the containment atmosphere were not considered in this calculation due to the heavy shielding around the core spray room.

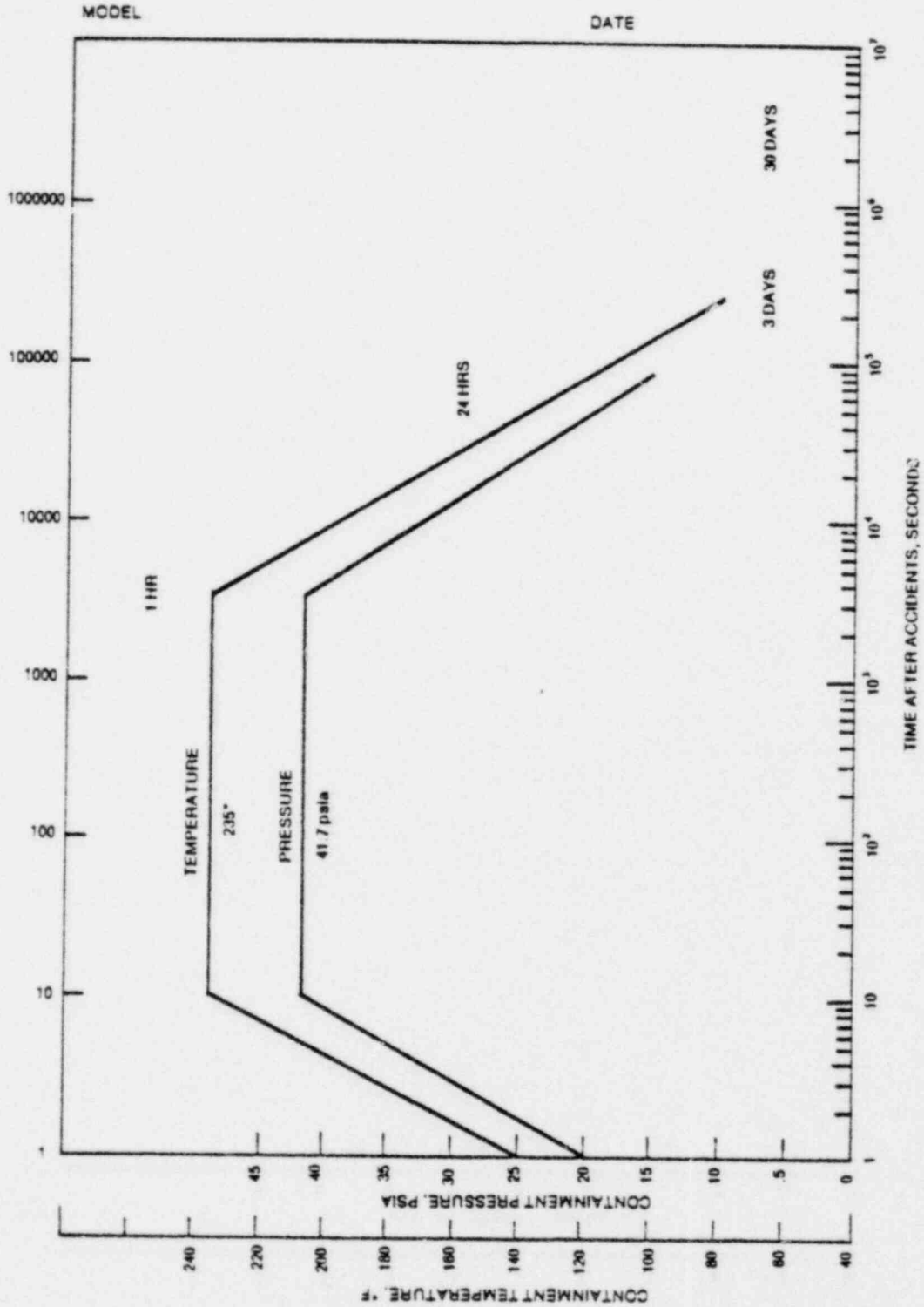


FIGURE SUPPLIED BY THE LICENSEE

Figure A-1. Time After Accidents, in Seconds Temperature and Pressure as a Function of Time After a LOCA

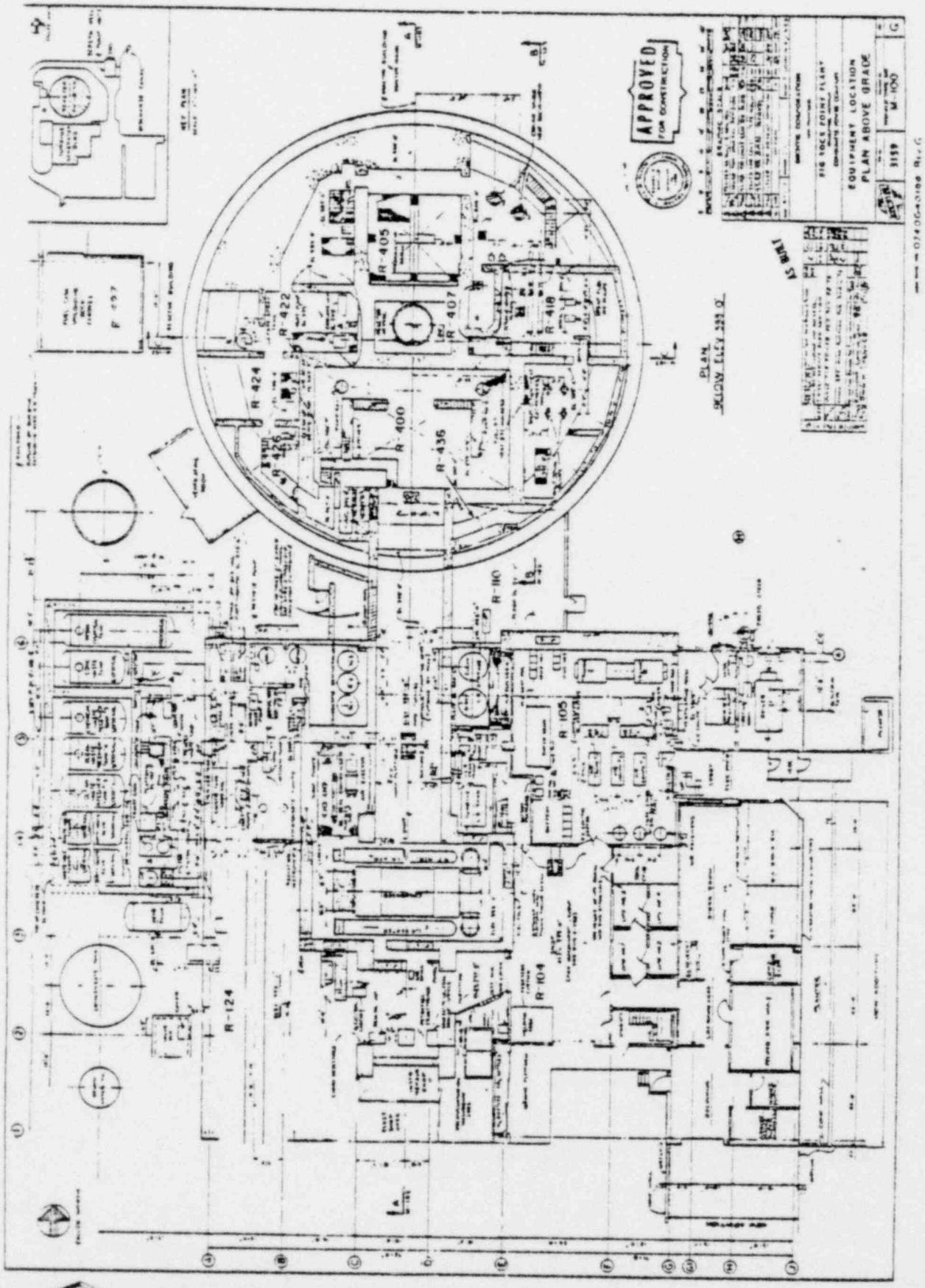


FIGURE SUPPLIED
BY THE LICENSEE

Figure A-2

DELETED MATERIAL IS PROPRIETARY INFORMATION

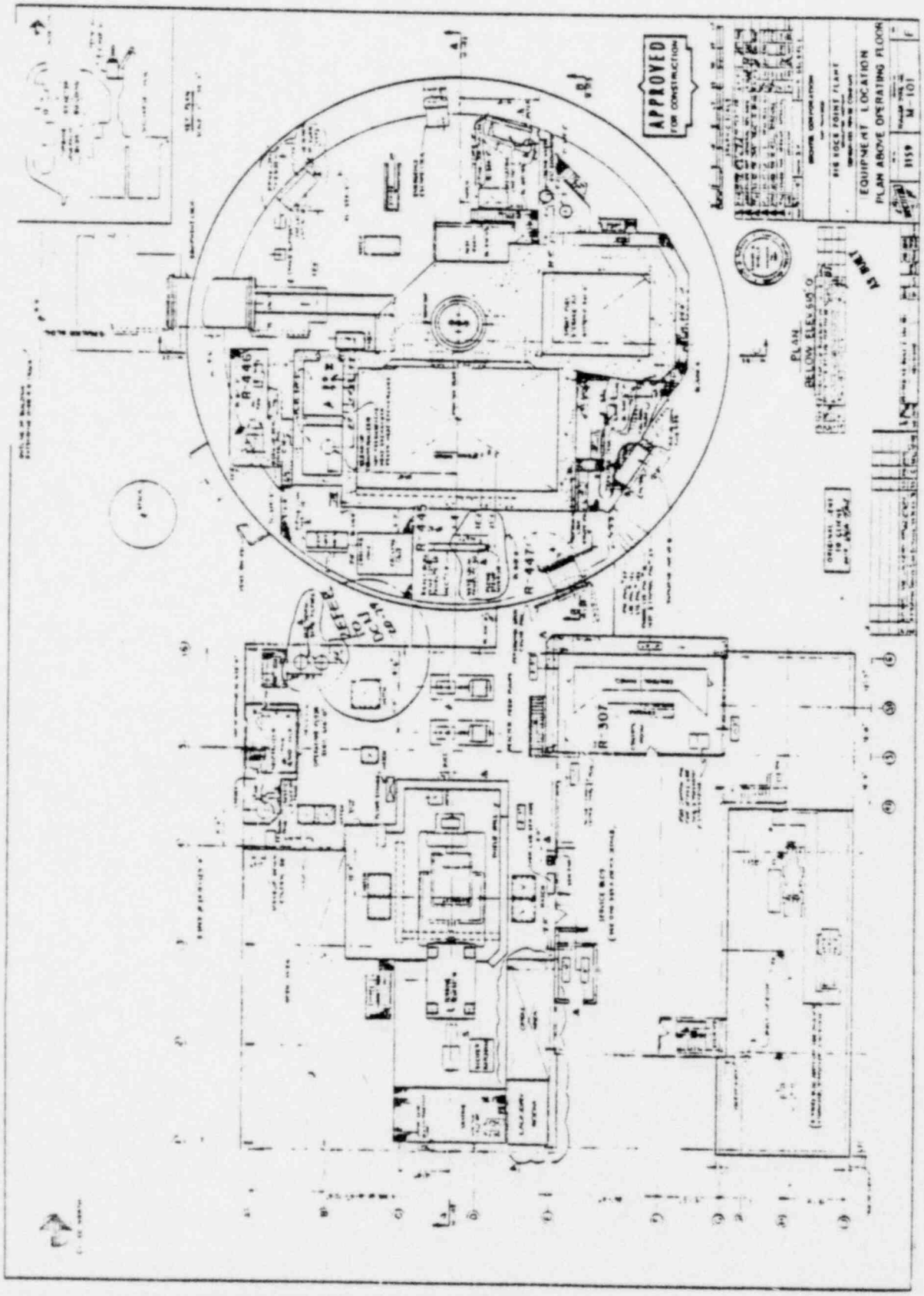


FIGURE SUPPLIED BY THE LICENSEE

Figure A-3

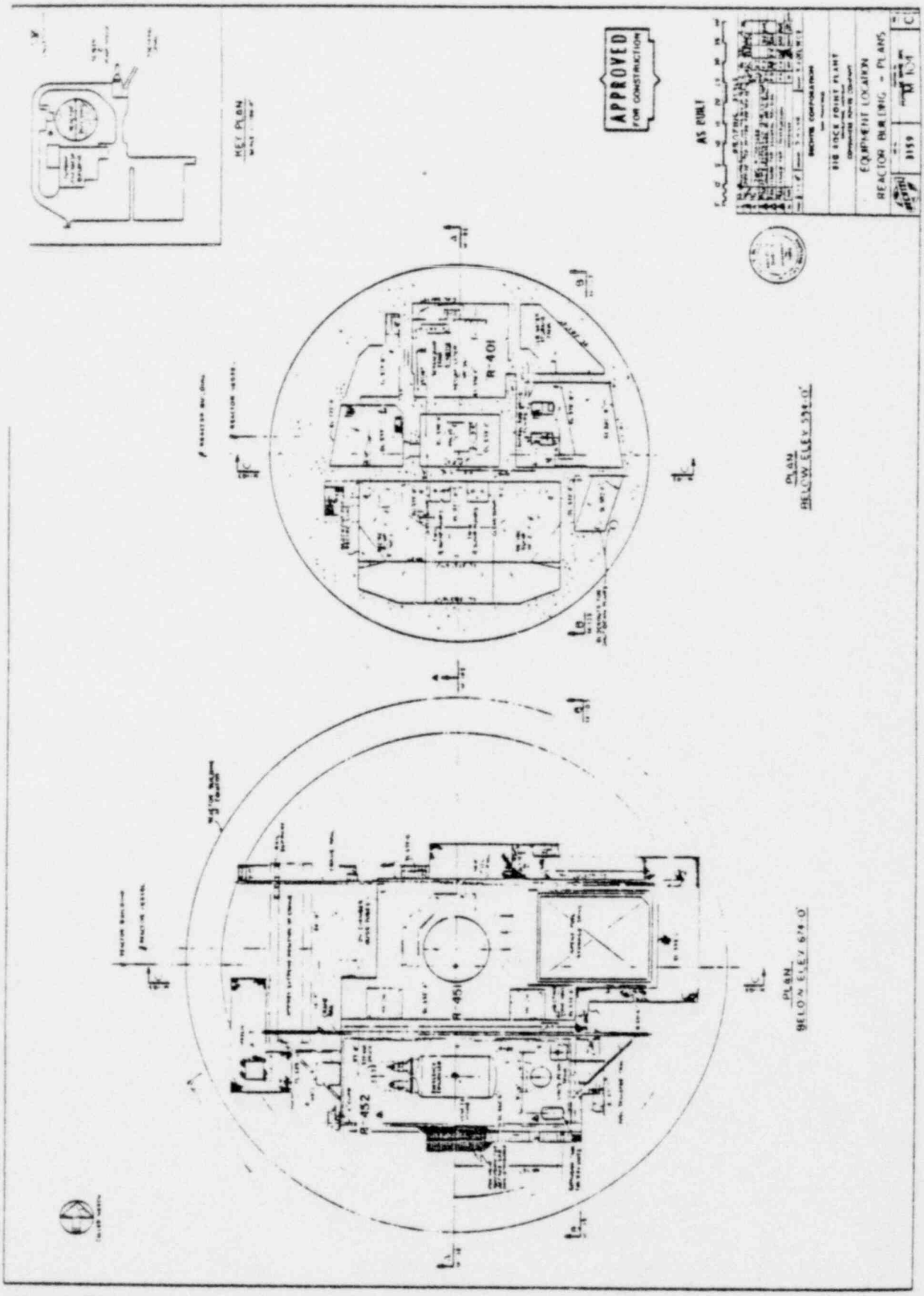
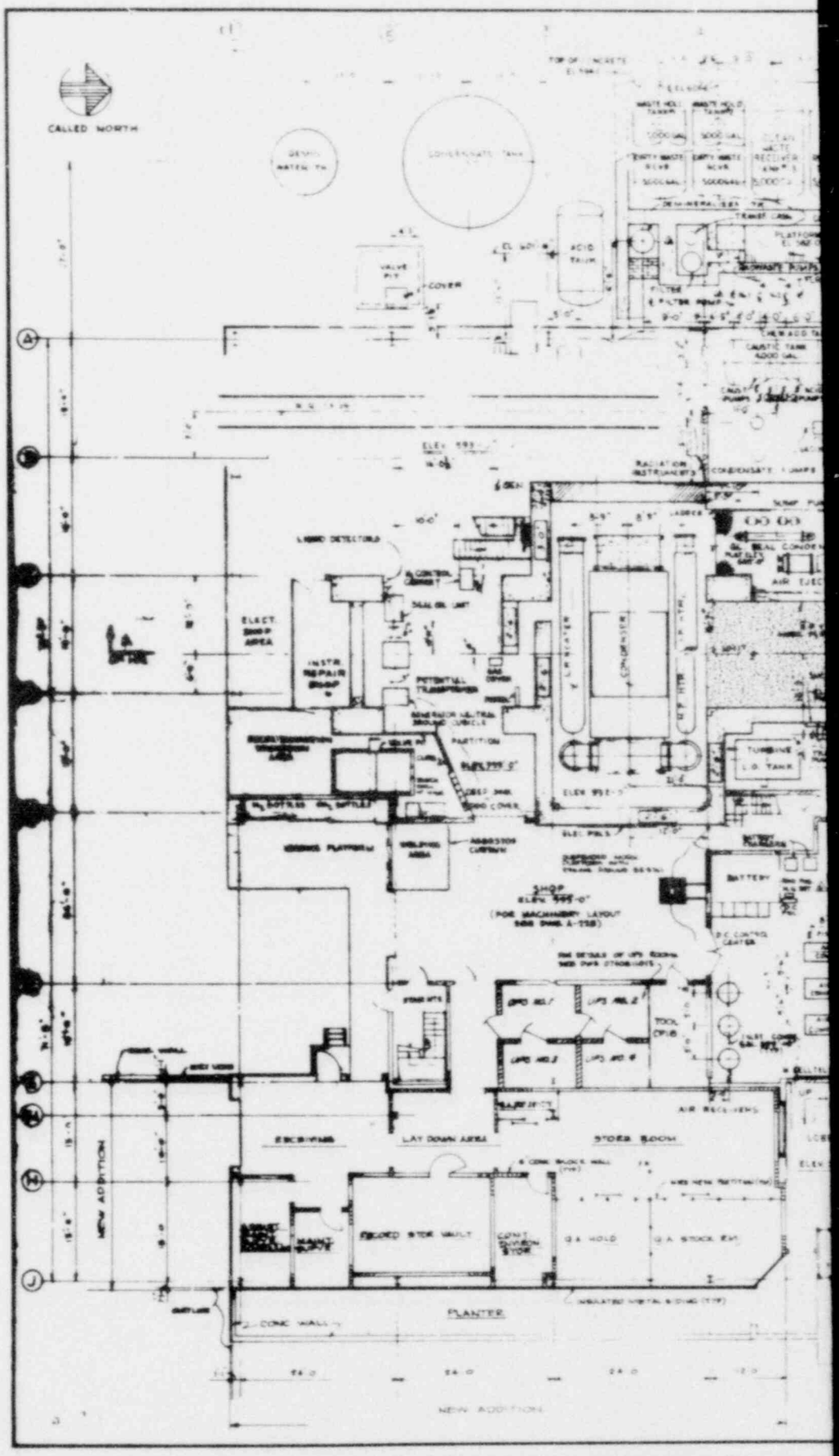


Figure A-4

POOR ORIGINAL



POOR ORIGINAL

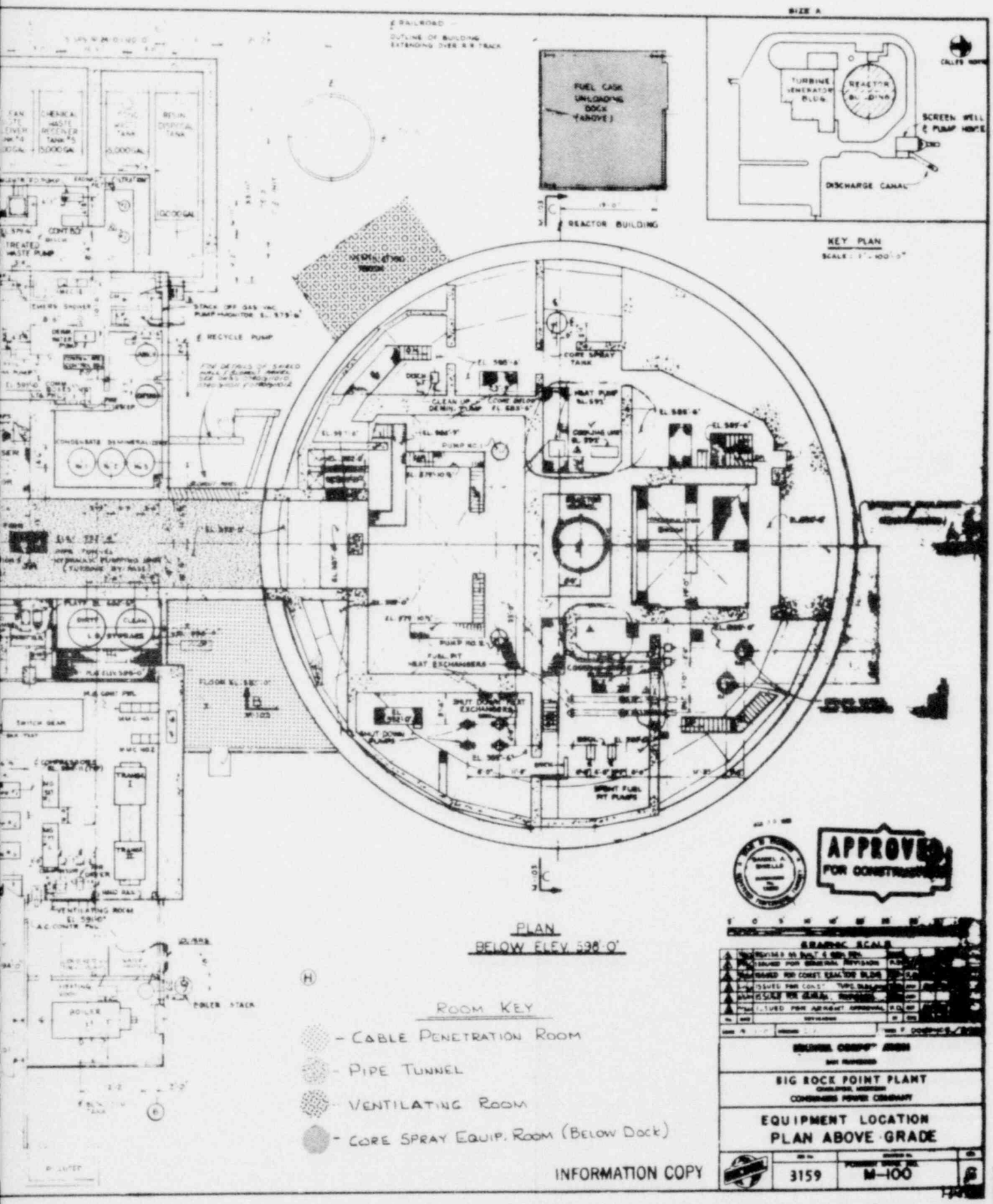


FIGURE SUPPLIED BY THE LICENSEE

APPENDIX B - EQUIPMENT ITEM TABLE

The following table lists safety-related electrical equipment for the Big Rock Point nuclear power plant. Equipment Item numbers provided in the table are used in the Equipment Qualification Documentation Review Summary Forms in Section 5 of this report and in the equipment qualification discussions presented in the body of the report.

This table was generated from a list of equipment items provided by the Licensee in Reference 90. FRC has grouped those plant equipment items that have identical qualification references, environmental service conditions, and functional time requirements, using the information provided by the Licensee.

PLANT: BIG ROCK POINT

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	LICENSEE SUBMITTAL PAGE REFERENCES	QUALIFICATION REFERENCES
1	Electrical Penetration GE Types 1-7	Containment	Long (30 days)	42	1,2,36,78,80, 4-5,4-65,4-66
2	Electrical Penetration Conax Corp. Types 10-12	Containment	Long (30 days)	49	3,5,60, 4-22,4-27,84, 2
3	Electrical Penetration Amphenol-Borg Type 8	Containment	Long (30 days)	51	36,37,79,4-67, 4-68,4-69,1
4	Electrical Penetration Manufacturer Not Stated Type 9	Containment	---	58	None
5	Flow Transmitters ITT-Barton 386 FT-2161 through FT-2164	Containment	30 days	60	61,33,62,63, 38,4-2,4-13
6	Level Switches Jo-Bell, Type R LS-3561 through LS-3565	Containment	Short (21 hrs.)	63	64,18,19, 65,81, 4-2
7	Level Switches Yarway 4320 PE LS-RE06A&B LS-RE20A&B	Containment	Short (10 min.)	66	13,14,15,66, 92.4,18,4-18
8	Level Switches Yarway 4420C LS-RE09 A-H	Containment	Short (1 hr.)	71	13,10,39,67,68, 92.4,4-18,15, 4-2, 4-14, 4-16

DELETED MATERIAL IS PROPRIETARY INFORMATION

PLANT: BIG ROCK POINT

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	LICENSEE SUBMITTAL PAGE REFFRENCES	QUALIFICATION REFERENCES
9	Level Transmitter ITT-Barton 386 LT-3171	Containment	Long	76	33,38,61,62,63
10	Level Transmitters Westinghouse 59DP4C997050 LT-3180 through 3183 LT-3184 through 3187	Containment	Long (30 days)	79-81	34,41,4-31
11A	Motorized Valve Actuators Limitorque SMA-00 Core Spray (MO-7051,-7061)	Containment	Long (30 days)	83	32,4-2,4-12, 4-15,6
11B	Motorized Valve Actuators Limitorque SMA-00 Containment Spray (MO-7068)	Containment	Long (30 days)	96	32,4-2,4-12, 4-15
11C	Motorized Valve Actuators Limitorque SMA-2-60 MSIV (MO-7050)	Containment	Long (30 days)	84	32,4-2,4-12, 4-15,6
12A	Motorized Valve Actuator Rotork 14A Containment Spray (MO-7064)	Containment	Long (30 days)	87	32,16,69,17,6, 4-2,4-12,4-15
12B	Motorized Valve Actuators Backup Core Spray (MO-7070,-7071)	Containment	Long	98	32,16,69,17,6, 4-2,4-12,4-15

PLANT: BIG ROCK POINT

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	LICENSEE SUBMITTAL PAGE REFERENCES	QUALIFICATION REFERENCES
13	Motorized Valve Actuator Limitorque SMA-00 Post-Incident Heat Exchanger Cooling Water (MO-7066)	Core Spray Room	Long (30 days)	91	32,4-12
14	Motorized Valve Actuators Limitorque SMA-1 Turbine Bypass Isolation (MO-7067)	Turbine Pipe Tunnel	Short (10 min.)	93	12,7,6,43
15	Motorized Valve Actuator Rotork 14A Core Spray HX (MO-7072)	Core Spray Room	Long (30 days)	102	32,4-46 4,4-12,4-15
16	Motor Starter GE CR109 for MO-7066 (#13)	Post-Incident Room	Long (30 days)	104	None
17	Electric Motors GE SK4364XJ1A11 P2A, P2B	Core Spray Room	Long (30 days)	106	70
18A	Pressure Switch ITT-Barton 289A PDIS-7814	Core Spray Room	Short (21 hrs.)	108	4-39,10
18B	Pressure Switch Mercoid, DAW-23-153 PS-638	Core Spray Room	Short (21 hrs.)	108	4-39,10

PLANT: BIG ROCK POINT

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	LICENSEE SUBMITTAL PAGE REFERENCES	QUALIFICATION REFERENCES
19A	Pressure Switch Static-O-Ring 12L-AA5-FSS PS-7064, 7064B	EP Room	V. Short (1 hr.)	110	44,4-44
19B	Pressure Switch Static-O-Ring 4NN-E411-YXSTT PS-664-667	EP Room	V. Short (1 hr.)	110	44,4-44
20	Pressure Switch Static-O-Ring 9TA-S4-11SSX12 PS-IG11A through IG11H	Containment	V. Short (1 hr.)	113	71,20,43,45,44, 73,4-24,4-23, 4-62,4-25
21	Pressure Transmitter Rosemount 1151GP PT-173, 174, 187	EP Room	Long (30 days)	117	46,4-32
22	Pressure Transmitter Rosemount 1152GP9A92PB PT-IA07C	Containment	Long (30 days)	119	21,4-29,47
23	Radiation Monitor Technical Assoc. CP-MU	EP Room	Long (30 days)	121	36,48,43
24A	Solenoid Valve ASCO HTX-800C61RF SV-4876	Containment	Long (30 days)	124	22,4-2,4-1, 4-20,4-52, 4-19,4-63, 4-64
24B	Solenoid Valve ASCO 830060RF SV-4869-4891	Containment	Long (30 days)	125	22,4-2,4-1, 4-20,4-52, 4-19,4-63, 4-64

PLANT: BIG ROCK POINT

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	LICENSEE SUBMITTAL PAGE REFERENCES	QUALIFICATION REFERENCES
25	Solenoid Valve ASCO 831620 (SV-4879)	Containment	Long (30 days)	128	4-1,4-2,4-20, 4-19,4-52,4-63, 4-64
26	Solenoid Valve ASCO 83006DR Isolation (SV-4832)	Containment	Long (30 days)	131	None
27	Solenoid Valve ASCO HTX-8300C61RF 830060R SV-4895, 4896, 4922	Turbine Bldg. Turbine Bldg.	Long (30 days) Long (30 days)	133 134	49 49
28	Solenoid Valve ASCO 830060R SV-4897	Turbine Bldg.	Long (30 days)	136	None
29	Solenoid Valve ASCO 830060R SV-4899, -4916	Turbine Bldg.	Long (30 days)	138 139	49
30	Solenoid Valve ASCO HTX831677 SV-4980 through 4983	Containment	Long (30 days)	141	4-1,4-2,4-20, 4-19,4-52,4-64, 4-63
31	Solenoid Valve Target Rock 73V001 SV-4984 through 4987	Containment	Short (1 hr.)	143	23,4-8
32A	Solenoid Valve ASCO 8316C44 SV-9151-9152	Sphere Vent. Room	Long (30 days)	146	49

PLANT: BIG ROCK POINT

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	LICENSEE SUBMITTAL PAGE REFERENCES	QUALIFICATION REFERENCES
32B	Solenoid Valve ASCO FT-8316D44 SV-9151-9154	Sphere Vent. Room	Long (30 days)	147	49
33	Solenoid Valve ASCO WP-80306 SV-9155, 9156	Sphere Vent. Room		149	49
34	Solenoid Valve ASCO 831622 SVNC-22F,G,H,J	Containment	Long (30 days)	151	4-1,4-2,4-19, 4-20,4-52,4-63, 4-64
35	Solenoid Valve ASCO 831622 SVNC-27A2A through SVNC-27F5B (64 Total) SVNC-22A,B,C,D	Containment	Short (10 min.)	153 154 155	50,4-1,4-2,4-19, 4-20,4-52,4-64, 4-63
36	Junction Box Manufacturer Not Stated JB-160-161,-164, -166,-167,180,170, 170A,B,171,171A,B; IG11A,B,C,D	Containment	Short (1 hr.)	158	4-2,4-4,4-26
37	Splice Insulation 3M No Model No.	Inside/ Outside Containment	Long (30 days)	160	24,82
38A	Terminal Blocks GE CR-151	Containment	Long (30 days)	164	75,36
38B	Terminal Blocks States Co. NT	Containment	Long	167	75,36

PLANT: BIG ROCK POINT

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	LICENSEE SUBMITTAL PAGE REFERENCES	QUALIFICATION REFERENCES
39A	Terminal Blocks Westinghouse 542247	Containment	Long (30 days)	167	25,42,40,76
39B	Terminal Blocks Westinghouse 805432	Containment	Long	167	25,42,40,76
40	Terminal Blocks GE EB-25	Containment	Long (30 days)	170	26,76
41	Terminal Connections AMP Special Products PIDG	Containment	Long (30 days)	173	27,4-6
42	Electrical Cable Anaconda 32277	Containment	Long (30 days)	175	29
43	Electrical Cable Cerro No model no.	Containment	Long (30 days)	178	28,29,30,51,8
44	Electrical Cable Kerite FR	Containment	Long (30 days)	182	31,72,74 4-9,4-11,4-10
45	Electrical Cable Raychem SLPE-Flamtrol	Containment	Long (30 days)	186	35,4-9,4-10, 4-11,4
46	Electrical Cable Raychem Flamtrol	EP Room	Long (30 days)	189	35,4-9,4-10, 4-11
47	Electrical Cable Rockbestos Firewall III	Sphere Vent. Room	Long (30 days)	192	52,91,44,46

PLANT: BIG ROCK POINT

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	LICENSEE SUBMITTAL PAGE REFERENCES	QUALIFICATION REFERENCES
48	Electrical Cable Manufacturer Not Stated	Containment	Long (30 days)	194	53,57,58
49	Electrical Cable Manufacturer Not Stated	Containment	Long (30 days)	197	54
50	Electrical Cable Manufacturer Not Stated	Containment	Long (30 days)	201	56,59,83
51	Electrical Cable Manufacturer Not Stated	EP Room	Long (30 days)	205	55,57,58
52	Electrical Cable G.E.	Containment (Inside Penetration)	Long (30 days)	208	56,9,83
53	Electrical Cable Okonite	Containment (Inside Penetration)	Long (30 days)	213	56,9,83,1
54	[Electrical Penetration Rundel]	[Containment]	[Long (30 days)]	[14 DITER]	[4-5, 4-65, 4-66]
55	[Cable Splices AMP Certiseal 324549, 324990]	[Containment]	[Long (30 days)]	[31 DITER]	[4-36, 4-55, 4-59, 4-61]
56	[Junction Box Rumsey Electric TB-240, TB243]	[Containment]	[Long (30 days)]	[16 DITER]	[4-[illegible], 4-33]

Note: Bracketed equipment items have been deleted from the Licensee submittal.

PLANT: BIG ROCK POINT

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	LICENSEE SUBMITTAL PAGE REFERENCES	QUALIFICATION REFERENCES
57	[Pressure Transmitter Foxboro E11GM-HSAE-1]	[Containment]	[Long (30 days)]	[22 DITER]	[4-28]
58	[Terminal Blocks Crouse-Hinds F17C222]	[Containment]	[Long (30 days)]	[32 DITER]	[4-14, 4-[illegible]3]
59	[Motor Operated Valve Limitorque SMA-000-5]	[Containment]	[Not cited]	[9 DITER]	[4-2, 4-12, 92.4,59]
60	[Cable Splices Manufacturer and Model Not Stated]	[Containment]	[Long (30 days)]	[36 DITER]	[4-61]
61A	Terminal Block Stanwick Co. Model Not Stated	Containment	Long (30 days)	None	None
61B	Splice Manufacturer and Model Not Stated	Containment	Long (30 days)	None	None

Note: Bracketed equipment items have been deleted from the Licensee submittal.

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

By generic letter, the NRC has transmitted to all SEP plants, including Indian Point 2 and 3 and Zion 1 and 2, DOR Guidelines for evaluating Class 1E equipment qualification and guidance for identifying equipment which must be qualified. In particular, guidance for identifying equipment is provided by NRC document "Guidelines for Identification of that Safety Equipment of SEP Operating Reactors for Which Environmental Qualification is to be Addressed."

The Licensee has submitted a list of safety-related systems required to function to mitigate the consequences of a design basis accident.

During the week of September 7, 1980, NRC and FRC reviewed the Licensee listing and the NRC listing at the plant site. As a result of these discussions and a CPC letter dated September 22, 1980, the following list represents the minimum number of systems and components for which NRC has determined that qualification is to be addressed.

Safe Shutdown Systems

Reactor Protection System*
Emergency Power AC-DC*
Emergency Condenser*

Accident Mitigating Systems (LOCA, MSLB, FWLB)

Containment Isolation
Core Spray (and Backup System)
Enclosure Spray (and Backup System)

* Required for both safe shutdown and accident mitigation.

Safeguards Activation
Radiation Monitoring and Sampling
Reactor Depressurization
Main Steam Isolation
Post-Incident System Long-Term Cooling
Fire Water System
Radwaste System (Isolation Valve Only)

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

Reactor Vessel Level
Reactor Pressure
Core Spray Flow**
Emergency Condenser Level
Containment Enclosure Spray Flow**
Fire System Strainer Differential Pressure**
Steam Drum Pressure
Steam Drum Level
Fire System Pressure**

**Required for accident mitigation only.

APPENDIX D - TABLE OF PHENOLIC RESINS

	Specific Gravity	Specific Heat	Thermal Conductivity (c.g.s. units) $\times 10^{-4}$	Thermal Expansion Coeffc. (per °C) $\times 10^{-5}$	Water Absorption* (mg)
Cast Resin	1·28-1·32	0·4-0·5	3-5	3-9	2-20
Moulding Material					
Wood-flour-filled	1·3-1·4	0·25-0·36	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	5-8	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

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

Table 3.3 MECHANICAL PROPERTIES OF RESINS

	Ultimate Tensile Strength (lbf/in^2) $\times 10^3$	Bending Strength (lbf/in^2) $\times 10^3$	Ultimate Shear Strength (lbf/in^2) $\times 10^3$	Ultimate Compression Strength (lbf/in^2) $\times 10^3$	Modulus of Elasticity (in tension) (lbf/in^2) $\times 10^3$	Modulus of Rigidity (in torsion) (lbf/in^2) $\times 10^3$	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	8-15	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	0·1-3·0
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	1,000-1,000		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:

Ogorklewicz, R. M. and Ritche, P. D. Phd.
Phenolic Resins. LONDON ILIFFE Books Ltd 1967.

APPENDIX E - EQUIPMENT ITEM CROSS-REFERENCE LIST

EQUIPMENT ITEM NO.	DRAFT INTERIM TECHNICAL EVALUATION REPORT SECTION	FINAL TECHNICAL EVALUATION REPORT SECTION
1	3.3.3.1	4.3.1.1
2	3.3.2.1	4.5.2.1
3	Not applicable	4.6.1
4	3.3.2.2	4.6.2
5	3.3.3.5	4.6.3
6	3.3.1.1	4.7.1
7	3.3.2.3	4.6.4
8	3.3.2.4	4.5.2.2
9	3.3.3.6	4.6.6
10	3.3.3.7	4.5.2.3
11A	Not applicable	4.5.2.4
11B	3.3.2.5	4.5.2.5
11C	3.3.3.7	4.5.2.4
12A and B	3.3.3.8	4.5.2.21
13A	3.3.2.6	4.3.1.2
13B	3.3.2.7	4.3.1.2
14	3.3.3.2	4.3.3.1
15	3.3.3.3	4.3.1.3
16	3.3.3.4	4.7.2
17	3.3.3.13	4.5.2.6
18A and B	3.3.3.10	4.7.7
19A and B	3.3.3.11	4.5.2.19
20	3.3.3.16	4.6.18
21	3.3.3.14	4.6.19
22	3.3.1.4	4.3.1.4
23	3.3.1.3	4.7.8
24A and B	3.3.3.15	4.5.2.7
25	3.3.3.9	4.5.2.8
26	3.3.3.1	4.5.2.9
27	3.3.1.2	4.7.3
28	3.3.1.2	4.7.4
29	3.3.3.12	4.5.2.10
30	Not applicable	4.5.2.11
31	Not applicable	4.5.2.12
32A and B	Not applicable	4.7.5
33	Not applicable	4.7.6
34	Not applicable	4.5.2.20
35	Not applicable	4.5.2.13
36	Not applicable	4.4.1
37	Not applicable	4.5.2.14
38A and B	Not applicable	4.6.5

EQUIPMENT ITEM NO.	DRAFT INTERIM TECHNICAL EVALUATION REPORT SECTION	FINAL TECHNICAL EVALUATION REPORT SECTION
39A and B	Not applicable	4.6.5
40	Not applicable	4.6.5
41	Not applicable	4.3.1.5
42	Not applicable	4.5.2.15
43	Not applicable	4.5.2.16
44	Not applicable	4.5.2.17
45	Not applicable	4.5.2.18
46	Not applicable	4.3.1.6
47	Not applicable	4.3.1.8
48	Not applicable	4.6.7
49	Not applicable	4.6.8
50	Not applicable	4.6.10
51	Not applicable	4.6.9
52	Not applicable	4.6.11
53	Not applicable	4.6.12
54	Not applicable	not used
55	Not applicable	4.6.13
56	Not applicable	4.6.14
57	Not applicable	4.6.15
58	Not applicable	4.6.16
59	Not applicable	4.4.2
60	Not applicable	4.6.17
61A and B	Not applicable	4.6.20

APPENDIX F - FRC EVALUATION OF LICENSEE - JUSTIFICATION FOR REMOVING EQUIPMENT FROM EEQ LISTING

NOTICE

Acceptance of Licensee justification for interim plant operation with unqualified equipment is beyond the scope of this Technical Evaluation Report. However, at the request of the NRC, FRC's evaluations of these justifications have been included. The evaluations are based upon FRC's judgment and experience, using the technical information available to FRC from the Licensee. They are intended to be used by the NRC as additional input to the judgments which are being made by the NRC in the Safety Evaluation Report.

In Section II.B of Reference 1, the Licensee provided justifications for removing certain equipment from the EEQ listing for Big Rock Point. This appendix provides FRC's evaluations of the Licensee's positions relative to this equipment.

1. MO-7069 Motorized Valve Actuator

LICENSEE POSITION:

This valve is electrically disabled in open position.

FRC EVALUATION:

Valves which are electrically disabled in their safety position do not require environmental qualification because potential failure of the valve can not prevent performance of the safety function and also cannot cause electrical faults to other safety-related components, circuits, or power supplies. Where this technique is used, specific plant procedures must be in effect to ensure that the valve will remain disabled during plant operations, in the hot-standby condition, and any other time when it may be relied upon to perform its safety function.

FRC CONCLUSION:

FRC finds that this valve was properly deleted from the listing provided that appropriate plant procedures are in place.

2. MO-7065

LICENSEE POSITION:

This valve is for the main steam line drain. It is a normally closed valve.

FRC EVALUATION:

This valve is the main steam line drain isolation valve inside containment. The main steam line drain is a 1.5-in line branching off the main steam line inside of the main steam line isolation valve. Although it is a normally closed valve, should it be opened at any time during plant operations, such as startup or heatup, it may be the only valve available to isolate an otherwise un-isolable steam leak directly from the reactor steam drum.

FRC CONCLUSION:

The Licensee's position provides a basis for interim plant operation; however, this valve should be environmentally qualified for its post-accident environment. It should not be deleted from the listing.

3. Containment Isolation Valve Limit Switches (Total of 20 Switches)

LICENSEE POSITION:

NRC staff, during their site visit, stated that limit switches used for indication only need not be qualified at this time. All of these position switches are associated with containment isolation valves. The emergency procedure requires "subsequent" operator action to first check the vent valves closed and later to verify automatic isolation valves closed. If they don't close, the operator is to attempt to close them manually. This manual attempt probably would be done by the S-5 switch ("close penetrations and scram").

FRC EVALUATION:

Limit switches which provide indication upon which post-accident operator action is required cannot be considered to be "indication only" devices.

Indication which provides the operator with information that safety-related equipment has performed its design function must be provided. Containment isolation valves provide a barrier to fission product release to the surrounding environment and the proper operation of these valves must be ensured by the operator. Even if the operator were to attempt manually closing of these valves, either with the S-5 switch or by other means, without proper indication there is no way for the operator to determine that one or more valves has failed or otherwise disabled and is not performing its safety function.

FRC CONCLUSION:

The Licensee's position provides a basis for interim plant operation; however, containment isolation valve limit switches must be qualified for the post-accident environment. They should not be deleted from the listing.

4. SV-4917

LICENSEE POSITION:

This solenoid valve controls isolation valve CV-4107 which is downstream of the main steam drain valve, MO-7065, which is normally closed. Since MO-7065 is normally closed, this valve is not required.

FRC EVALUATION:

As discussed in Item No. 2 above, should valve MO-7065 be open at any time when isolation of the main steam drain line is required, the functioning of both MO-7065 and CV-4107 would be required. In the event of a single active failure to MO-7065, CV-4107 would be relied upon to isolate this line which is directly connected to the reactor steam drum. In addition, in a post-accident scenario, both valves are relied upon to be shut and are tested in accordance with 10 CFR 50, Appendix J to ensure that post-accident containment leakage is maintained below acceptable limits.

FRC CONCLUSION:

The Licensee's position provides a basis for interim plant operation; however, this valve should be qualified for its post-accident environment. It should not be deleted from the listing.

5. SV-NC22C and D

LICENSEE POSITION:

These solenoid valves control the scram dump tank isolation valves. They are excluded from qualification because their failure to function during a LOCA would allow leakage to the reactor building sumps, a negligible problem compared to the LOCA.

FRC EVALUATION:

The scram dump tank is initially vented to atmosphere. Upon a scram, the vent/drain valves are shut to isolate the tank. When the scram stroke is completed, the accumulator pressures continue to hold the rods in the reactor. If the accumulator charging system does not function, internal shuttle valves open and apply reactor pressure under the rod drive pistons until the dump tank is filled and system pressures are equalized. At this point, the weight of the rods are supported by locking mechanisms which are engaged to hold the rods at the fully inserted position. Should valves CV-NC-11 and/or CV-NC-12 remain open following a scram or reopen at some later time due to the failure of solenoids SV-NC-22C and SV-NC-22D, a continuous leakage path would exist from the reactor vessel to the reactor building sumps throughout the entire post-accident period.

FRC CONCLUSION:

The Licensee's position provides a basis for interim plant operation; however, these solenoid valves should be qualified for the post-accident environment. They should not be deleted from the listing.

6. I & C Power Panel 2Y

LICENSEE POSITION:

This panel is located in containment. It is fed from the 2B bus through I & C transformer 2B. A breaker located in the station power room, 22Y1, provides protection to the 2B bus and the other I & C preferred panels (1Y and 3Y) in the event 2Y fails.

FRC EVALUATION:

The Licensee indicates that in the event of failure to panel 2Y because of non-environmental qualification, power supply bus 2B and other preferred panels (1Y and 3Y) would be protected by breaker 22Y1. The Licensee has not, however, provided a determination that the loss of panel 2Y, particularly considering possible single-active-failure to components redundant to safety-related components powered by 2Y, is acceptable from an accident mitigation standpoint.

FRC CONCLUSION:

Power Panel 2Y should not be deleted from the listing until a determination is made that loss of the panel has no impact upon accident mitigation.

7. RE-8258, RE-8259

LICENSEE POSITION:

These radiation monitoring instruments cause the containment vent valves to close due to radioactivity in the atmosphere. The intent originally was to have the valves close on high radiation in the event of a fuel handling accident rather than a LOCA. When thinking in terms of a LOCA, these devices become a backup to any of the scram input signals since any scram closes the vent valves. Pertinent scram inputs for this argument are low steam drum level or containment high pressure. In any larger size break LOCA, the radiation instruments will not have a chance to respond prior to the high containment pressure switches or the low steam drum level switches. In the very small size LOCA, the radiation levels may not be high enough to cause a vent valve closure via the radiation instruments. No credit has been taken for these instruments in mitigating the consequences of a small steam line break.

FRC EVALUATION:

Despite any anticipatory signals which will close the containment vent valves upon a scram, high radioactivity in the containment must be protected against, with regard to the position of the containment vent valves. Short of an exhaustive study of the effects of small break LOCA on various monitored parameters, including the possibility of a single active failure to monitoring equipment, there is no apparent justification for failing to qualify the radiation monitoring instruments for the post-accident environment.

FRC CONCLUSION:

The Licensee's position provides a basis for interim plant operation; however, these instruments should be qualified for the post-accident environment. They should not be deleted from the listing.

8. Acoustic Monitoring of Steam Drum Reliefs (Total of 12 Monitors)

LICENSEE POSITION:

This equipment is used to provide acoustic monitoring on the steam drum relief valves. This equipment is presently scheduled to be qualified by Babcock and Wilcox. Also, since this equipment is used for monitoring only, it can be qualified at a later time.

FRC EVALUATION:

These monitors provide the operator with information that safety-related equipment is performing its design function. The Licensee has indicated that qualification by Babcock and Wilcox is presently scheduled.

FRC CONCLUSION:

This equipment should be qualified and it should not be deleted from the listing so that eventual qualification may be properly documented.

APPENDIX G - BIG ROCK POINT PLANT CHEMICAL SPRAY

Licensee has submitted conflicting information on the composition of the spray to be applied during a LOCA. The submittal of October 31, 1980 states that lake water alone is to be used. A letter [92.1] in response to questions raised at the meeting of September 10, 1980 states that sodium pentaborate concentrate is to be added to the spray. FRC believes the low concentration and modest pH of the borate spray, determined as described below, result in very low potentials for corrosion with the exception of aluminum alloys. Most aluminum alloys are subject to pitting corrosion in dilute borate solutions. The Licensee should clarify the spray composition, and if borates are used, Licensee should identify equipment exposed to the spray which contains or is contained within aluminum alloys and present evidence of its qualification for use with borate sprays. Certain equipment items were tested with borated sprays which are not identical to the Licensee's spray, but FRC believes the differences in composition are acceptable since the Licensee's spray is more dilute than those used in testing. These items include cable manufactured by Anaconda, Cerro and Kerite; cable splices manufactured by Raychem, terminal blocks manufactured by Westinghouse; terminal connections manufactured by AMP Special Products; and level transmitters manufactured by Westinghouse. Many items are contained within sealed housings or cabinets. These are of unstated materials; they cannot be considered qualified. Items so contained include flow transmitters, level switches, pressure switches, pressure transmitters, and certain solenoid valves.

The letter presents the following data: concentrate volume 850 gallons, flooded containment volume 34,800 cubic feet, concentration 19 to 30 weight percent sodium pentaborate. From this FRC calculates an equilibrium sodium pentaborate concentration of 0.07 to 0.13 weight percent. The pH of these solutions could not be found in the usual compilations, i.e., CRC Handbook of Chemistry and Physics or International Critical Tables. Two manufacturers of

borates, Kerr-McGee, Oklahoma City, OK, and U.S. Borax, Anaheim CA, were contacted for pH information, with results as tabulated below:

Concentration, wt.%*		pH at 20°C
Kerr-McGee	U.S. Borax	
-	0.85	8.48
1	-	8.4
-	2.1	8.01
3	-	8.0
-	4.1	7.59
-	5.1	7.4
6	-	7.6
9	-	7.2

* $Na_2B_{10}O_{16} \cdot H_2O$

Neither company had pH information at the lower concentrations which the Licensee proposes to use at the LOCA temperatures. By extrapolation, values of about 9 would be expected.