TECHNICAL EVALUATION REPORT

EQUIPMENT ENVIRONMENTAL QUALIFICATION

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT UNIT NO. 3

NRC DOCKET NO. 50-286

NRC TAC NO. 42474

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC TASK 206

Prepared by

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

FRC Group Leader: C. J. Crane

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

NRC Lead Engineer: J. Lombardo

April 9, 1981

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The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

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SRC = Synergic Resources Corporation

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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 EVALUATION

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

1.2 GENERIC ISSUE BACKGROUND

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

Qualification criteria applied during the licensing of the older nuclear power plants have been modified over the years, and specific industry standards concerning qualification have been revised as the design of reactor systems has changed and as regulatory and operating experience has accumulated. Examples of such standards are IEEE Standards 279-71, 323-74, 382-74, 317-76, 334-74, 381-77, 382-80, and 627-80. NRC NUREG documents 0413 and 0588 have been developed to address this topic. In particular, NUREG-0588 (published for comment in December 1979) formally presented the NRC staff positions regarding selected areas of environmental qualification of safety-related electrical equipment in the resolution of General Technical Activity A-24, "Qualification of Class IE Safety Related Equipment." The positions documented therein are applicable 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 power plants deviated from current licensing criteria. The subject of electrical equipment environmental qualification (SEP Topic III-12) was selected for accelerated evaluation as part of this program. Seismic qualification of equipment was to be addressed as a separate SEP topic. In December 1977, the NRC issued a generic letter to all SEP plant licensees requesting that they initiate reviews to determine the adequacy of existing equipment qualification documentation.

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

On May 31, 1978, the NRC Office of Inspection and Enforcement issued IE Circular 78-08, "Environmental Qualification of Safety-Related Electrical

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

The review of the licensee responses indicated certain deficiencies in the scope of equipment addressed, definition of harsh environments, and adequacy of 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 IE Electrical Equipment in Operating Reactors" [12].* (The document is hereafter referred to as the "DOR Guidelines.") The document was prepared as a screening standard for reviewing all operating plants, including SEP plants. It was originally intended that the licensees evaluate their qualification documentation in accordance with the DOR Guidelines. However, initial NRC review of this documentation, which was compiled to support licensee submittals, revealed the need for obtaining independent evaluations and for accelerating the qualification review program.

In October 1979, the NRC awarded Franklin Research Center (FRC) a contract to provide assistance in the "Review and Evaluation of Licensing Actions for Operating Reactors," which included an assignment for review of equipment environmental qualification documentation under SEP Topic III-12. FRC was to review equipment environmental qualification documentation and to

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

present the results in the form of a Technical Evaluation Report for the 11 oldest plants (included in the SEP review).

On January 14, 1980, the NRC Office of Inspection and Enforcement issued the DOR Guidelines and IE Bulletin 79-01B, which expanded the scope of IE Bulletin 79-01 and requested additional information on environmental qualification of safety-related electrical equipment at operating facilities, excluding the 11 facilities undergoing the SEP review. This Bulletin cited the DOR Guidelines as the criteria to be used in evaluating the adequacy of the safety-related electrical equipment qualification. The scope of the review was expanded to include high energy line breaks (inside and outside containment) in addition to equipment aging and submergence. The NRC advised the licensees that the criteria contained in the DOR Guidelines would be used in its review of licensee submittals; problems arising from this review would be resolved using NUREG-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 which included a second document, "Guidelines for Identification of That Safety Equipment of SEP Operating Reactors for Which Environmental Qualification Is To Be Addressed" [12], 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 [15], specifying that licensees and applicants must meet the requirements set forth in the DOR Guidelines and NUREG-0588 regarding environmental qualification of safety-related electrical equipment in order to satisfy 10CFR50, Appendix A, General Design Criteria, Section I, Criterion 4. This Order also established that the Safety Evaluation Reports on this subject, to be prepared by the NRC staff, must be issued on February 1, 1981 and that all subsequent actions to be taken by licensees to achieve full compliance with the DOR Guidelines or NUREG-0588 must be completed no later than June 30, 1982.

1.3 SPECIFIC ISSUE BACKGROUND

By a letter dated March 5, 1980, the Power Authority of the State of New York (PASNY) was formally notified by the NRC that the review of environmental qualification for safety-related electrical equipment for the Indian Point Unit No. 3 nuclear power plant would be conducted under SEP Topic III-12. Information requested from PASNY included identification of the electrical equipment required to perform safety functions while subjected to design basis accident environments. Definitions of environmental service conditions at equipment locations and the status of environmental qualification were also requested. In addition, documentation pertaining to qualification was to be compiled and organized for review by NRC and FRC. In response to this request, PASNY provided information via a submittal transmitted by letter dated April 28, 1980 [1]. In March 1980, NRC representatives visited the Indian Point Unit No. 3 plant site to make a preliminary determination of adequacy of equipment environmental qualification and to discuss the identification and evaluation of safety-related equipment. On July 1 and 2, 1980, NRC and FRC representatives visited the Indian Point Unit No. 3 plant site, inspected safety-related systems and components, and discussed the April 28, 1980 submittal with PASNY representatives. PASNY submitted additional information by letters dated July 30 and August 11, 1980.

FRC issued a Draft Interim Technical Evaluation Report to NRC on September 9, 1980 [16]. Copies of the report were transmitted to PASNY by the NRC.

On August 29 and September 19, 1980, NRC notified PASNY that all supplemental information on equipment environmental qualification must be submitted by November 1, 1980.

On October 30, 1980, additional responses and qualification information were provided by the Licensee [9, 10].

1.4 SCOPE OF THE EVALUATION

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

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

2. NRC CRITERIA FOR ENVIRONMENTAL QUALIFICATION

2.1 CRITERIA PROVIDED BY THE NRC

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

- "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" [12]
- "Guidelines for Identification of That Safety Equipment of SEP Operating Reactors for Which Environmental Qualification Is To Be Addressed" [12].

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 [13], "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 profiles as functions of time. If containment spray is to be used, the impact of the spray on required equipment should be assessed.

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

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

Equipment submergence (inside or outside containment) should be addressed where the possibility exists that submergence of equipment may result from HELBs or other postulated occurrences. Supplement 2 to IE Bulletin 79-01B [13] 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 [13] 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 [13] 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 [14] permits the submittal of qualification documentation regarding the TMI Action Plan equipment and the equipment required to achieve and maintain a cold shutdown condition to be delayed as follows:

- "Qualification information for installed TMI Action Plan equipment must be submitted by February 1, 1981.

- Qualification information for future TMI Action Plan equipment (ref. NUREG-0737, when issued), which requires NRC pre-implementation review, must be submitted with the pre-implementation review data.

- Qualification information for TMI Action Plan equipment currently under NRC review should be submitted as soon as possible.

- Qualification information for TMI Action Plan equipment not yet installed which does not require pre-implementation review should be submitted to NRC for review by the implementation date.

- The qualification information for equipment required to achieve and maintain a Cold Shutdown condition ... will 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 [13] provides the following additional criteria:

> "Sequential testing requirements are specified in NUREG-0588 and the DOR Guidelines. Licensees must follow the test requirements of the applicable document.
>
> 1. If the test has been completed without aging in sequence, justification for such a deviation must be submitted.
>
> 2. If testing of a given component has been scheduled but not initiated, the test sequence/program should be modified to include aging.
>
> 3. Test programs in progress should be evaluated regarding the ability to comply by incorporating aging in the proper sequence. These would then fall in the first or second category."

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

Supplement 2 to IE Bulletin 79-01B [13] provides the following additional criteria:

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

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

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

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

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

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

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

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

3. METHODOLOGY USED BY FRC

The Licensee, Power Authority of the State of New York, listed an extensive number of safety-related electrical equipment items* in various locations of Indian Point Unit No. 3 in its submittals to the NRC [1,6,9]. 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 reduced the list to 61 different equipment items that formed the basis for the review. In this report, the term "equipment item" refers to a specific type of electrical equipment, designated by manufacturer and model, which is representative of all identical equipment in a plant area exposed to the same environmental service conditions (e.g., Flow Transmitter, Fischer & Porter, Model 10B2496, located within containment). Appendix A contains the environmental service conditions for each location, Appendix B contains the tabulation of the equipment items, and Appendix C lists the plant systems and display instruments identified by the Licensee and the NRC as being essential to safety.

Using the list of safety-related electrical equipment items, FRC reviewed each equipment item 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) on September 9, 1980 [16]. Qualification data summary forms used to summarize salient data compiled from the various information sources were included in the DITER.

In developing the present final Technical Evaluation Report (TER), FRC used the DITER and the Licensee submittals and qualification reference documents [1,2,3,6,9,10]. This information was analyzed by FRC to determine:

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

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

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

NRC CATEGORIES AND DEFINITIONS

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

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

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

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

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

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

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

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

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

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

- NRC Category III
  EQUIPMENT THAT IS EXEMPT FROM QUALIFICATION

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

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

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

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

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

- NRC Category V
  EQUIPMENT THAT IS UNQUALIFIED

The DOR Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology and program be referenced and made available for review of all Class 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: (1) documentation reflecting qualification as specified in the DOR Guidelines has not been made available for review, (2) the documentation is inadequate, or (3) the documentation indicates that the equipment item has not successfully passed the required tests.

- NRC Category VI
  EQUIPMENT FOR WHICH QUALIFICATION IS DEFERRED

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

4. TECHNICAL EVALUATION

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

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

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

All equipment item* numbers are associated with the information presented in References 1 and 9.

4.1 METHODOLOGY USED BY THE LICENSEE

The Licensee's submittal [1] contains a brief discussion of the basic approach and methodology used in preparing the data and information submitted. The review by FRC has generated the following observations and comments.

4.1.1 COMPLETENESS OF EQUIPMENT LIST

The Licensee has opted to defer the qualification review of electrical equipment associated with (i) TMI Lessons Learned and (ii) cold shutdown requirements, in accordance with Section 2.2.5 of this report. Also, the

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

Licensee has chosen to include only equipment subjected to severe service conditions as a result of postulated accidents. The qualification review of the equipment located where it is subjected to "mild" environmental service conditions also is deferred (see Section 2.2.3), and the Licensee has not identified the equipment in this category. In addition, the Licensee has neither fully defined what is regarded as "severe service conditions" nor fully justified the selection of the plant areas considered to have a "mild" environment.

The following equipment may be subjected to "harsh" environments, but was not included in the Licensee's equipment list [1]; it was identified in the DITER [16] although not addressed in the Licensee's response.

- o sensor, wiring, and controls in the system that senses a HELB in the auxiliary pump room and acts to isolate the steam supply line (see the discussion in Section D.1 of Appendix D)
- o cables and cable splices, terminal blocks, and connectors located outside of containment
- o temperature detector, pressure switch, terminal blocks, and splices associated with the hydrogen recombiner.

FRC assumes that other equipment items not listed in References 1 and 9 (e.g., inverters, motor control centers, and switchgear) are located in areas of the plant that do not experience "harsh" environments.

4.1.2 ENVIRONMENTAL SERVICE CONDITIONS

4.1.2.1 TEMPERATURE AND PRESSURE INSIDE CONTAINMENT

The Licensee has used the temperature/pressure profile curves based on a large-break LOCA as the "worst-case" accident for evaluating the qualification of equipment in this program. The NRC independently assessed the short- and long-term temperature profiles within containment [11] and stated that the Licensee's temperature/pressure profiles are acceptable for the purposes of this accelerated environmental qualification review. The NRC also calculated somewhat higher peak conditions (268°F/44 psig) and suggested that the Licensee should note that some margin in its qualification effort would be prudent.

4.1.2.2 TEMPERATURES IN LOCATIONS OUTSIDE CONTAINMENT

As noted previously, the Licensee has not fully justified the temperature conditions claimed for the various areas outside containment. For the present, except for the auxiliary pump room and the steam and feedline penetrations area, FRC has based this EEQ review on the temperature conditions claimed by the Licensee and given in Appendix A of this report. If a subsequent review discloses that the temperature conditions in various locations are more severe, (e.g., because of HVAC system limitations), further evaluation of the equipment in these locations will be required. In particular, the Licensee states that the temperatures in all areas outside of containment are "ambient" or "negligible change," even under HELB conditions.

The Licensee acknowledges that a short-term temperature transient will occur in the auxiliary pump room and the steam and feedline penetrations area. For the former location, this transient is quantified as being only 5 minutes in duration. As explained more fully in Section D.1 of Appendix D, FRC does not agree with the stated conditions because they are based on the proper functioning of equipment for which qualification has not been established. For the latter location, the Licensee has stated that the only change from normal environmental conditions in this location as the result of a HELB is a 0.42 psi pressure increase, based on a large break leading to a rapid pressure rise that causes the metal siding on the building to be blown open. The Licensee has not provided any evidence that the temperature increase is negligible for other than large break events. The HELB conditions cited for other operating PWR plants reviewed by FRC have been based on saturated steam conditions existing for periods of up to an hour or more. For this review, therefore, FRC has used a temperature condition of 213°F for 20 minutes for both the auxiliary pump room and steam and feedline penetrations area. (This corresponds to accidents involving other than large breaks, in which the maximum pressure conditions stated by the Licensee are reached relatively slowly.) The Guidelines require that qualification be demonstrated under accident conditions for a period of one hour plus the expected operating time.

4.1.2.3 NUCLEAR RADIATIONS

The radiation dose levels within containment cited by the Licensee in Reference 1 are based on gamma radiation alone and do not include the beta contribution. The Licensee has not established that the beta radiation is reduced to insignificant levels (either by shielding or by other means for all of the equipment). Also, as noted in the DITER, Reference 7 (prepared for Indian Point Unit 2) lists higher dose levels for many equipment items. The Licensee [9] takes exception to FRC's reference to a report prepared for another plant; however, the response fails to address the question regarding the correct choice of dose levels on which to base the EEQ review. (FRC has not implied that the values in Reference 1 are wrong, only that the discrepancy in values should be addressed.)

4.1.3 AGING AND QUALIFIED LIFE

The Licensee has stated:

"An aging and qualified life program is ongoing at Indian Point No. 3. All the suppliers have been contacted and given the model numbers and serial numbers of their supplied equipment. The vendors are supplying bills of material for this equipment. When the bills are received, the material is reviewed and the effects of radiation and thermal aging is determined from the data that is available for the material."

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

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

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

Establishing the qualified life for equipment is a technically challenging task because of the paucity of information concerning the degradation of materials and components under long-term exposure to the environmental service conditions of a nuclear power generating station. As discussed more fully in Reference 17, with the possible exception of certain simple materials, there is no rigorous basis for establishing equipment qualified lifetimes for periods approaching an installed lifetime of 40 years. Furthermore, applicable information regarding possible long-term synergistic effects of temperature, humidity, nuclear radiations, etc., is extremely limited.

In accordance with the Guidelines in this program, the licensees are required to establish a qualified life for equipment subject to thermal and radiation aging. In addition, surveillance, maintenance, and replacement programs should be established for equipment that may be subject to age-related degradation. The licensees should review the qualified life values and the present installed life of the equipment to determine a replacement schedule for each equipment item (or subcomponents thereof). As noted above, these schedules may be revised as new information becomes available.

4.2 EQUIPMENT QUALIFIED FOR PLANT LIFE

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

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

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

For Indian Point Unit No. 3, no equipment falls within this category.

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

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

For Indian Point Unit No. 3, no equipment falls within this category.

4.3 EQUIPMENT QUALIFIED WITH RESTRICTIONS

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

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

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

4.3.1.1 Equipment Item No. 18A
Solenoid Valves Located in the Pipe Penetrations Area
Automatic Switch Co. (ASCO) Model NP-8316
Actuates Containment Pressure Relief Valves (PCV-1191, 1192)
(Licensee Reference 2.8)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.9):

The Licensee's reference is a proprietary test report describing a qualification program conducted for a number of ASCO valves. FRC's review of this report has resulted in the following conclusions:

a. Of the valve models tested, the one with a model number that most closely matches that of the installed equipment is sample No. 6,

solenoid enclosure, and normally closed operation. The Guidelines require that the test specimen be the same as the equipment being qualified. The Licensee did not present information describing the installed item; a statement that it is identical to the test sample;

or an analysis comparing the impact of deviations between the test specimen's specific design features, materials, and production procedures and those of the installed equipment. Therefore, an independent conclusion cannot be reached regarding the extent to which the results of the test program provide valid evidence of qualification. The Licensee should provide certification that the important features of the installed equipment are the same as those of the test specimen.

b. The environmental and operational service condition parameters used in the qualification test program exceeded those dictated by plant-specific requirements in all cases except the of the steam temperature/pressure profile. This deficiency is not regarded as being significant. The Licensee submittal did not consider the nuclear radiation dose resulting from (i) normal plant operations and (ii) beta radiation (including the bremsstrahlung radiation it creates while being attenuated). However, the test program included a sufficiently large gamma radiation dose (Mrd) that the other dose contributions can be considered to have been accommodated.

c. The pre-aging simulated in the test program was intended to represent an installed life (and hence a qualified life) of ambient temperature. The ambient temperatures at the installed locations within the plant are lower; hence, the qualified life is longer. The Licensee should explicitly determine the qualified life and establish a replacement schedule for the item if this lifetime is less than the period for which the plant is licensed to operate.

LICENSEE RESPONSE:

The concern is that the ASCO solenoid valve NP 8316A75E, which was installed, differs from the tested model NP 831665E. The difference between the valves is the size of the pipe connection and the orifice. The concern of aging is on-going; however, since we have data to indicate that the solenoid will perform its function for a minimum of 4 years, a small replacement schedule is incorporated. This schedule will be modified as necessary when more data on aging is received.

FRC EVALUATION:

As the Licensee has more fully identified the equipment model number, FRC agrees that the cited reference is valid for this equipment item. In the discussion of the cited reference in connection with Equipment Item No. 18B,

Because the Licensee states there is neither steam nor spray nor submergence environment in the pipe penetrations area, this failure mode is not a concern for this equipment item. FRC interprets the Licensee Response to indicate agreement that at present the qualified life is 4 years.

FRC CONCLUSION:

This equipment item is assigned to NRC Category II.a because adequate evidence of qualification has been presented. The qualified life was identified as 4 years by the Licensee.

4.3.1.2 Equipment Item Nos. 28A and 28B
Limit Switches Located Within Containment (28A) and in the Pipe Penetrations Area (28B)
NAMCO Model EA-180
(Original Licensee Reference 2.9; Final Reference 10.2)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.1.3):

The referenced qualification report is for a NAMCO Model EA-740-20000 limit switch, which is not the same as the installed equipment. FRC's evaluation of the status of qualification for this item follows:

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. Therefore, an independent conclusion cannot be reached regarding the extent to which similarity exists, and the validity of the cited test as evidence of qualification has not been established.

b. The Licensee has stated that this item is used for position indications only or that a failure of the limit switch will not cause the valve to change positions and is not required to perform a safety function. In addition, the air supply to the valves located within containment, on which some of these switches are mounted, is removed during the accident. If the NRC concurs with the Licensee's position, the equipment can be considered exempt from qualification requirements.

LICENSEE RESPONSE:

FRC's concern is the comparison of the NAMCO EA-180 limit switch to the EA-740. Enclosed is the qualification report from NAMCO Controls covering the EA-180 limit switch.

The switches are installed at Indian Point No. 3 with Conax connectors. The connector is a sealed unit that was tested and qualified for use inside containment. The connectors were thermally aged prior to being radiated to 150 Mrads, LOCA tested to 340 (max.) and seismically tested to 3 g's vertical and 3.5 g's horizontal. The LOCA test consisted of a chemical spray consisting of Boron and Na^{O}H with a 10.5 pH continued for 24 hours before switching to demineralized water. After the 30-day test, all results were satisfactory, and no observable deficiencies were detected.

FRC EVALUATION:

Although these limit switches provide position indication only, the function of the switches is basically to provide the operator with an indication of proper valve positioning. This is important because lack of indication that the valve is closed could initiate an inappropriate action on the part of the operator (refer to Sections D.5 and D.6 of Appendix D.)

The test report [10.2] supplied by the Licensee demonstrates that the conditions to which these limit switches would be exposed are enveloped by the test conditions.

The use of the Conax connector is considered satisfactory to seal the switch intervals against steam entry as required by the manufacturer.

FRC CONCLUSION:

This equipment item is assigned to NRC Category II.a. The Licensee should establish a conservative qualified life (see Section 4.1.3). (Note: For FCV 1170 and 1172, FRC concurs with the Licensee that position indication is not required. See Section D.5 of Appendix D.)

4.3.1.3 Equipment Item No. 33
Limit Switches Located Within Containment
NAMCO Model EA-740
(Licensee Reference 2.9)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.1.8):

FRC has reviewed the referenced qualification report and notes the following:

a. The test specimen was a NAMCO Model EA-740-20000 limit switch, which may be different from the model installed in the plant. Also, the body of the limit switch used in the test was sealed to ensure that no steam entered the switch during the LOCA test. The Licensee should ascertain that the electrical connections and covers of the installed switches are sealed in the same manner. If not, the referenced test report is not valid as evidence of qualification. The Guidelines require that the test specimen must be the same as the equipment being qualified. The Licensee did not present an analysis comparing the impact of deviation between the test specimen's specific design features, materials, and production procedures and those of the installed equipment. Therefore, an independent conclusion cannot be reached regarding the extent to which the two units are similar, and the validity of the test as evidence of qualification is inconclusive.

b. The Guidelines require that the possibility of degradation of the materials used in the equipment be explicitly considered and that a qualified life period be established. This has not been done.

c. The environmental parameters in the test program fully envelop those for the installed equipment. This provides qualitative confidence that the equipment will perform adequately.

d. The Licensee has stated that this item is used for position indication only and that the air supply is removed from the valve to which the switch is attached during the accident. If the NRC agrees that the Licensee's position is valid, the equipment can be considered exempt from qualification requirements.

LICENSEE RESPONSE:

The switches are installed at Indian Point No. 3 with Conax connectors. The connector is a sealed unit that was tested and qualified for use inside containment. The connectors were thermally aged prior to being radiated to 150 Mrads, LOCA tested to 340 (max.) and seismically tested to 3 g's vertical and 3.5 g's horizontal. The LOCA test consisted of a chemical spray consisting of Boron and NaOH with a 10.5 pH continued for 24 hours before switching to demineralized water. After the 30-day test, all results were satisfactory, and no observable deficiencies were detected.

Conax connectors were also used on the switches in Section 3.3.1.3 [4.3.1.2 of this report].

FRC EVALUATION:

FRC has reviewed the Licensee Response and subsequent information identifying the Conax connectors as type N11001-33. The Licensee has demonstrated that the NAMCO EA-740 limit switch is sealed to prevent entry of steam into the limit switch. As noted in the DITER, type testing was performed and established qualification to DOR Guidelines. However, no qualified life was established as noted in FRC's DITER comments.

FRC CONCLUSION:

This equipment item is assigned to NRC Category II.a. The Licensee should establish a conservative qualified life (see Section 4.1.3).

4.3.1.4 Equipment Item No. 44
Connectors Located Within Containment
Conax Model N11001-33
(Final Licensee Reference 10.4)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

None

LICENSEE STATEMENT (made in connection with Equipment Item No. 33):

> The switches are installed at Indian Point No. 3 with Conax connectors. The connector is a sealed unit that was tested and qualified for use inside containment. The connectors were thermally aged prior to being radiated to 150 Mrads, LOCA tested to 340 (max.) and seismically tested to 3 g's vertical and 3.5 g's horizontal. The LOCA test consisted of a chemical spray consisting of Boron and NaOH with a 10.5 pH continued for 24 hours before switching to demineralized water. After the 30-day test, all results were satisfactory, and no observable deficiencies were detected.
>
> Conax connectors were also used on the switches in Section 3.3.1.3 [4.3.1.2 of this report].

FRC EVALUATION:

The Licensee has identified the connectors as Conax type N11001-33 and stated that Report IPS-409 [10.4] applies. From review of the documentation referenced by the Licensee, FRC has the following comments and conclusions:

a. With regard to chemical spray, the test report describes testing with and spray, which fulfills the Guidelines requirement.

b. The manufacturer states in the test report that test data base configurations were established to qualify a series of conductor modules and that the materials are identical and configurations similar to production modules. Specifically, metal parts are ; sealants are ; conductor insulation is ; wire markers are ; and thermocouple-pair conductors are stated to be representative of module construction type connectors.

c. The report notes that age conditioning was performed to simulate normal operation. Radiation exposure to a dose was applied. Seismic tests and LOCA testing were performed as recommended in IEEE Stds 323-74 and 344-75. As recommended in IEEE Std 323-74, flame tests were performed and data presented to demonstrate that the modules performed satisfactorily. Flame and seismic tests are beyond the scope of this review. FRC agrees that the LOCA test and radiation exposure tests exceed the requirements for LOCA/HELB for Indian Point Unit No. 3. FRC concludes that the tests conducted exceed the Guidelines requirements.

d. The connector modules at Indian Point Unit No. 3 are Part No. N-11001-32, while Report IPS-409 refers to as the part number tested. However, Conax has stated that the series designation was changed to N-11000- series designation for standard identification and that both designations refer to the same conductor module assemblies. FRC therefore concludes that the installed conductor modules are the same as the type tested.

e. FRC notes in Table 5.3.1 of report IPS-409 that after aging, irradiation, and LOCA simulation, the insulation resistance increased, the helium leak rate increased, and the voltage used in the dielectric strength test was reduced so that the test values did not agree with the specification criteria of Table 3.2 of the test report. Although these data show some degradation, FRC concludes that the conductor modules are satisfactory to meet Guidelines requirements because the testing is more severe than required by the Guidelines and the test deviations described above would not affect the satisfactory performance of the equipment installed at Indian Point Unit No. 3.

FRC CONCLUSION:

This item is assigned to NRC Category II.a. The Licensee should establish a conservative qualified life for this item (see Section 4.1.3).

4.3.1.5 Equipment Item Nos. 37A and 37B
Electro-Pneumatic Pressure Transducers Located in Steam/Feedline Area and the Auxiliary Pump Room
Fisher Controls Company, Type 546
37A: AFW Flow Control Valves (FCV-405 A, B, C, and D; 406 A, B, C, and D)
37B: MS Atmospheric Relief (PCV-1134, 1135, 1136, and 1137)
(Licensee Reference 2.13)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.18):

This reference is an excerpt from a brief laboratory report that indicates the results of a steam exposure test in which no pre-aging was done. The Licensee has not stated the temperature that may result from a steam line break in the penetration area, so FRC has assumed a temperature profile in order to have a reasonable basis for making a quantitative evaluation. FRC also notes that Fisher Catalog indicates that the normal operating temperature for this device is to .

The results of FRC's review of the cited test report follow:

a. The environmental parameters used in the steam exposure test exceed the expected service conditions by a wide margin.

b. The Guidelines state that aging of test specimens is not required if the component does not contain materials known to be susceptible to significant degradation due to thermal or radiation aging. The materials used in this equipment have not been identified, and the period of qualified life has not been established.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

As previously stated, this equipment was tested in a steam environment by the manufacturer. The testing envelops the conditions identified in Appendix A of this report for the auxiliary pump room and steam/feedline area. The requirements of the DOR Guidelines are satisfied except for aging and qualified life (see Section 4.1.3).

FRC CONCLUSION:

This equipment item is assigned to NRC Category II.a. A conservative qualified life should be established (see Section 4.1.3).

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

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

4.3.2.1 Equipment Item No. 42A
Hydrogen Recombiner Panel Located in the Pipe Penetrations Area
Westinghouse Electric Corporation, Model Not Stated
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.23):

The Licensee states that this equipment is to be moved to an area with a less harsh environment, presumably because of limitations on personnel access [8], and that the control panel does not have materials with radiation exposure degradation thresholds lower than the specified value at the present location of this equipment. The Licensee did not provide a Bill of Materials to enable an independent verification to be made, and has not provided any information regarding aging degradation of the materials or the environmental parameters at the new location, as is required by the Guidelines. This information should be provided for review.

LICENSEE RESPONSE:

The hydrogen recombiner control panel area is to be shielded from this radiation source. This shielding will decrease the exposure in the area to 4.0×10^2 rads; hence, we consider this area non-hostile.

It is worth noting that NUREG-737 issued on October 31, 1980 will reduce the radiation expected in the area of the Hydrogen Recombiner Control Panel. Based on a complete analysis of the fields, we may not need to shield the piping below the panel which is the source of radiation. If this becomes the case, we will revise the Environmental Qualification Submittal to reflect change.

FRC EVALUATION:

The Licensee has responded that the panel will be provided with a radiation shield to reduce its exposure to 400 rd unless NUREG-737 relaxes the radiation qualification requirements for the panels. The equipment will not be subjected to a harsh environment during a LOCA or MSLB inside containment.

FRC CONCLUSION:

This equipment item is assigned to NRC Category II.b because the Licensee is committed to possible modification involving the shielding of the control panel from radiation in order to eliminate its harsh environmental service conditions during a postulated LOCA or MSLB inside containment. The Licensee has not provided a statement on the qualified life of the panel and this should be addressed (see Section 4.1.3).

4.3.2.2 Equipment Item No. 27
Solenoid Valves Located in the Pipe Penetrations Area
Lawrence Model 629BC85PS
Actuates Hydrogen Recombiner Isolation Valves (IV-2A, 2B, 3A, 3B, 5A, and 5B)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.13):

The Licensee states that these valves are required to function for an extended period following a LOCA, but that they do not contain materials susceptible to degradation at the nuclear radiation integrated dose levels expected at their mounting point. The Licensee has not identified the materials used, so an independent determination of this conclusion cannot be made. Also, the following Guidelines requirements have not been addressed: aging degradation has not been considered; qualified life has not been established; and a program has not been established to ascertain whether any in-service failures during the installed life of the equipment are the result of aging degradation.

LICENSEE RESPONSE:

> We are designing a shield for the source of radiation that will degrade the environment in the location of the Hydrogen Recombiner Control Panel. Therefore, the associated shielding analysis yielded a total integrated dosage less than initially reported in our April 28, 1980 submittal. The new dosage will be 400 rads. Hence we consider this area non-hostile.

FRC EVALUATION:

The proposed plant modification is satisfactory provided (i) it does reduce the radiation exposure by a factor of approximately 100, and (ii) it does not lead to overheating of the solenoids by restricting free air movement past them to dissipate heat. The Licensee has not made an assessment of aging degradation. The Guidelines require an assessment of aging degradation, considering the long-term exposure to temperature and humidity and, from this, an explicit determination of the qualified life on a conservative basis.

FRC CONCLUSION:

This equipment item is assigned to NRC Category II.b. The proposed addition of radiation shielding will eliminate the possibility of degradation induced as the result of exposure to nuclear radiations. The effect of the added shielding on the local ambient temperature and the qualified life should be established. The Licensee should determine the qualified life of non-metallic parts based on manufacturer's recommendations so that proper maintenance can be scheduled and performed.

4.3.2.3 Equipment Item No. 32B
Limit Switches Located in the Pipe Penetrations Area
Micro Switch Model EXHAR-3
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.1.7):

The stated design basis event environment deviates from ambient conditions only by a 30-day exposure of 0.02 Mrd accumulated dose. In Appendix C of the DOR Guidelines, the threshold level of radiation damage for materials typical of those used in a limit switch is 0.1 Mrad. In addition, the Licensee states that the limit switch is for position indication only. Provided that the NRC agrees that the valve position indication is not a safety function, FRC finds that qualification is not required for this item.

LICENSEE RESPONSE:

> We are designing a shield for the source of radiation which will degrade the environment in the location of the Hydrogen Recombiner control panel. This yields a decrease in the exposure to the limit switch to 400 rads. This is a significant change in the exposure to the equipment; hence, we consider this area non-hostile.

FRC EVALUATION:

These limit switches provide indication that the valves have closed upon receipt of a containment isolation signal. This information is significant in the mitigation of potential accidents in that the operators need indication as to whether or not containment isolation valves have performed their safety function. FRC agrees that shielding is a desirable modification to minimize radiation exposure.

FRC CONCLUSION:

This item is assigned to NRC Category II.b because the Licensee has committed to shield this item. The Licensee should establish a conservative qualified life for this item (see Section 4.1.3).

DELETED MATERIAL IS PROPRIETARY INFORMATION.

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

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

4.3.3.1 Equipment Item Nos. 1, 2, and 3
Motorized Valve Actuators Located Within Containment
1: Limitorque Model SMB-00 with Class H Motor Insulation
Actuates HH Injection Valves (MOV-856A through H, J, K);
Filter Dousing Valves (MOV-856A through H, J, K);
Recirculation Pump Discharge Valves (MOV-1802A, B);
RHR Isolation Valves (MOV-745A, B);
RHR Flow Control Valves (MOV-1869A, B)
2: Limitorque Model SMB-0 with Class H Motor Insulation
Actuates Recirculation Spray Valves (MOV-889A, B)
3: Limitorque Model SMB-3 with Class H Motor Insulation
Actuates RHR Exchanger Isolation Valves (MOV-746, 747, 899A, B)
(Licensee References 2.1, 2.2, 2.20, 2.21; Added Final Reference 10.1)

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

Reference 2.1 contains a test report for a Limitorque actuator with a motor (Class H insulation). Reference 2.2 is a test report for a Limitorque SMB-0 actuator with a Reliance motor (Class RH insulation). References 2.20 and 2.21 are letters that provide justifications for extrapolating the results from tests on units with Reliance motors to units with Peerless motors with Class H insulation. An attachment to Reference 2.21 indicates that a material evaluation was conducted, but the details of this evaluation were not made available. No conclusions can be drawn from these latter two incomplete references.

FRC has reviewed References 2.1 and 2.2 in detail and notes the following:

a. Although the test reported in Reference 2.2 was quite comprehensive (including thermal aging, load cycling, nuclear radiation, and 30-day steam/chemical spray exposure) and the test unit's performance was satisfactory, the validity of the cited test as evidence of qualification has not been established. 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. Therefore, an independent conclusion cannot be reached regarding the extent of similarity.

b. Also, although the test specimen used in the test reported in Reference 2.1 was the same as (or closely similar to) Equipment Item No. 2, it is obviously different than Item Nos. 1 and 3. In addition, the following deficiencies are noted:

   o The actuator was not subjected to thermal aging. The Guidelines state that thermal aging of test specimens is not required if the component does not contain materials known to be susceptible to significant degradation due to aging. The materials used in this equipment have not been identified.

   o 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 radiation, as is required by the Guidelines.

   o There were   of the unit during the test.

LICENSEE RESPONSE:

The main concern is that the equipment tested was not an exact duplicate of the equipment installed. In Report B0058 [10.1], which we believe is already in Franklin Research Center's possession, the subject of thermal aging of the actuator is addressed and shown to be acceptable. The subject of exposure to radiation of the actuator is also addressed in B0058 satisfactorily.

FRC EVALUATION:

Reference 10.1 (Limitorque Report B0058) provides a generic discussion of Limitorque's approach to qualification of its equipment. This report states:

> "The qualification of the Limitorque Size SMB-0, as reported in the documentation of each of the four tests, was used to generically qualify all sizes of Limitorque operators for the environmental test conditions in accordance with IEEE 382-1972. The Size SMB-0 actuator is an average mid-size unit, and all other sizes of the type SMB, SB, SBD AND SMB/HBC are also deemed qualified. All sizes are constructed of the same materials with components designed to equivalent stress levels, same clearances and tolerances with the only difference being in physical size which varies corresponding to the differences in unit rating."

For this equipment, FRC accepts the validity of the approach by Limitorque of qualifying a generic "family" of different size units based upon the tests of the mid-range size.

The Licensee has not assessed aging degradation. The Guidelines require that this be done, considering the long-term exposure to temperature and humidity. The Licensee should establish a conservative qualified life.

FRC CONCLUSION:

- This equipment item is assigned to NRC Category II.c because, while FRC believes qualification can be established for a period less than the licensed life of the plant, the Licensee has not obtained information from the manufacturer that confirms that the cited references apply to the equipment installed in the plant. Such confirmation should be obtained and the Licensee should analyze the aging data for the components of the equipment and from this establish a conservative qualified life (see Section 4.1.3).

4.3.3.2 Equipment Item Nos. 4A and 4B
Motorized Valve Actuators Located in the Pipe Penetrations Area (4A) and Safety Injection Room (4B)
Limitorque Model SMB-0 With Class B Motor Insulation
4A: Actuates Containment Sump Stop Valves (MOV-885A,B) and BIT Discharge Valves (MOV-1835A,B)
4B: Actuates BIT Injection Valves (MOV-1852A,B)
(Licensee References 2.1 and 2.19)

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

Reference 2.1 contains a test report for an actuator with a motor with Class B insulation. Reference 2.19 is a test report for an SMB-0 actuator with a Reliance motor with Class B insulation. The test sequence of Reference 2.1 is more severe than that of Reference 2.19, and is more severe than required to simulate the expected environments in the areas where this equipment is located.

Reference 2.19 reports a test which reflects current practice in qualification testing; therefore, it is preferred as evidence of qualification. The environmental parameters, aging considerations, and other aspects of the test program were more than adequate to demonstrate qualification for these equipment items. FRC therefore concludes that the requirements of the Guidelines have been satisfied.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

The Licensee has not established that the cited references are directly applicable to this equipment; this can be done only by obtaining a statement from Limitorque. However, from a general knowledge of this equipment and the fact that the Licensee states that only the radiation exposure increases significantly as a result of an accident, FRC believes that the Licensee will be able to demonstrate conclusively that this equipment is qualified.

FRC recommends that the Licensee review the vendor's data on aging for the electrical components in this equipment and make a conservative estimate of qualified life.

FRC CONCLUSION:

This equipment item is assigned to NRC Category II.c because, while FRC believes qualification can be established for a period less than the licensed life of the plant, the Licensee has not obtained information from the manufacturer that confirms that the cited references apply to the equipment installed in the plant. Such confirmation should be obtained and the Licensee should also analyze the aging data for the components of the equipment and from this establish a conservative qualified life (see Section 4.1.3).

4.3.3.3 Equipment Item Nos. 5A, 5B, 8, and 9
Motorized Valve Actuators Located in the Pipe Penetrations Area (5A and 9) and Safety Injection Room (5B and 8)
5A & 5B: Limitorque Model SMB-00 with Class B Motor Insulation
5A: Actuates RHR Exchanger CW Supply Valves (MOV-822A,B), RCP Seal Water Valve (MOV-222) HH Recirculation Valves (MOV-888A,B), and RCP CW Supply Valves (MOV-769,784,786,789,797; and FCV 625)
5B: Actuates SI Mini-Flow Valves (MOV-842, 843), HH SI Discharge Valves (MOV-851A,B), and RWST Discharge Valve (MOV-1810)
8: Limitorque Model SMB-000 with Class B Motor Insulation
Actuates SI Pump Suction Isolation Valves (MOV-887-A,B)
9: Limitorque Model SMB-1 with Class B Motor Insulation
Actuates RHR Isolation Valve (MOV-744)
(Licensee References 2.1 and 2.19)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.1.1):

Reference 2.1 contains a test report for an actuator with a motor with Class B insulation. Reference 2.19 is a test report for an SMB-0 actuator with a Reliance motor with Class B insulation. The test sequence of Reference 2.1 is more severe than that of Reference 2.19, and is more severe than required to simulate the expected environments of the pipe penetrations area.

Reference 2.19 reports a test which reflects current practice in qualification testing; therefore, it is preferred as evidence of qualification. FRC has the following comments with regard to this reference:

a. The environmental parameters, aging considerations, and other aspects of the test program were more than adequate to demonstrate qualification for these equipment items.

b. The test report is for a Limitorque model SMB-0-25 actuator with a Reliance motor with Class B insulation. Although the Guidelines require that the test specimen be the same as the equipment being qualified, FRC believes the differences in construction would not have affected the test results, because of the margins on the environmental parameters used in the test program.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

The Licensee has not established that the cited references are directly applicable to this equipment; this can be done only by obtaining a statement from Limitorque. However, from a general knowledge of this equipment and the fact that the Licensee states that only the radiation exposure increases as a result of an accident (all other environmental stresses are "nonharsh"), FRC believes the Licensee will be able to demonstrate conclusively that this equipment is qualified.

FRC recommends that the Licensee review the vendor's data on aging for the electrical components in this equipment and make a conservative estimate of qualified life.

The Licensee has not cited Reference 10.1 for this equipment, but has cited it for Equipment Item Nos. 1, 2, and 3. FRC believes the Licensee should cite this reference as evidence that the tests in the references originally cited are applicable to larger and smaller MVA sizes.

FRC CONCLUSION:

This equipment item is assigned to NRC Category II.c because, while FRC believes qualification can be established for a period less than the licensed life of the plant, the Licensee has not obtained information from the manufacturer that confirms that the cited references apply to the equipment installed in the plant. Such confirmation should be obtained and the Licensee should analyze the aging data for the components of the equipment and from this establish the qualified life (see Section 4.1.3).

4.3.3.4 Equipment Item No. 7
Motorized Valve Actuators Located Within Containment
Limitorque Model SMB-2 with Class B Motor Insulation
Actuates Accumulator Discharge Isolation Valves (MOV-894, A, B, C, D)
(Licensee References 2.1 and 2.19; Added Final Reference 10.1])

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

Reference 2.1 contains a test report for an actuator with a Electric Co. motor with Class B insulation. Reference 2.19 is a second test report for an SMB-0 actuator with a Reliance motor with Class B insulation. The test sequence of Reference 2.19 is not as severe as that required to simulate the expected environments within the containment. Therefore, the results of Reference 2.19 cannot be used as proof of qualification. With regard to Reference 2.1, FRC has the following comments:

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

b. The referenced test program consisted of high temperature functional tests, a steam exposure including interim functional testing, accelerated aging (stated to be equivalent to 40 years), seismic test, a 150 open/close life-cycle test, and a steam exposure that included boric acid spray.

As a result of this test program, Westinghouse (in Licensee Reference 2.1) stated that actuators with Class B motor insulation will be supplied "where only short term (less than 12 hours) operation is required." The Licensee has stated that the required operating time for these equipment items is less than 8 hours and that it is not normally used during an accident. The NRC should determine the acceptability of the Licensee's position. The Guidelines state that a failure at any time during a test should be considered inconclusive with regard to demonstrating the ability of the component to function for the entire period prior to the failure.

c. Licensee Reference 2.1 also states that another Class B motor (the motor manufacturer was not stated; from the serial number FRC deduces that this was a motor) was exposed to a radiation dose level of . No difference in response was detected between this unit and an identical one that was not irradiated. However, this similarity of response included overheating, production of smoke, and a transient open circuit during a severe test program involving 220-V operation (instead of 440 V, as in the plant), 45 forward and reverse (f/r) cycles while at room temperature, and 45 f/r cycles at ambient. In order to use this test to establish qualification for the installed units, the Licensee should provide an analysis demonstrating that the materials are the same in both the motor installed in the units in the plant and the motor that was irradiated to   in the test.

LICENSEE RESPONSE:

With respect to the replacement of the motor operators committed, Limitorque Report B0058 [10.1] again addresses the difference in the tested units and units which are to be installed.

FRC EVALUATION:

Reference 10.1 (Limitorque Report B0058) provides a generic discussion of Limitorque's approach to qualification of its equipment. This report states:

"The qualification of the Limitorque Size SMB-0, as reported in the documentation of each of the four tests, was used to generically qualify all sizes of Limitorque operators for the environmental test conditions in accordance with IEEE 382-1972. The Size SMB-0 actuator is an average mid-size unit, and all other sizes of the type SMB, SB, SBD and SMB/HBC are also deemed qualified. All sizes are constructed of the same materials with components designed to equivalent stress levels, same clearances and tolerances with the only difference being in physical size which varies corresponding to the differences in unit rating."

For this equipment, FRC accepts the validity of the approach by Limitorque of qualifying a generic "family" of different size units based upon the tests of the mid-range size.

The Licensee has not assessed aging degradation. The Guidelines require that this be done, considering the long-term exposure of this equipment to temperature, nuclear radiation, and other environmental parameters. The Licensee should establish a conservative qualified life.

As discussed in Section D.4 of Appendix D, FRC agrees with the Licensee's statement that the function of this equipment is accomplished early in the accident scenario.

FRC CONCLUSION:

This equipment item is assigned to NRC Category II.c because, while FRC believes qualification can be established for a period less than the licensed life of the plant, the Licensee has not obtained information from the manufacturer that confirms that the cited references apply to the equipment installed in the plant. Such confirmation should be obtained and the Licensee should analyze the aging data for the components of the equipment and from this establish a conservative qualified life (see Section 4.1.3).

4.4 NRC Category III
EQUIPMENT THAT IS EXEMPT FROM QUALIFICATION

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

4.4.1 Equipment Item No. 37C
Electro-Pneumatic Pressure Transducers Located in the Pipe Penetrations Area
Fisher Controls Company, Type 546
Water Return Valves TCV-1104 and 1105
(Licensee references not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.1.10):

Although the Licensee did not cite Reference 2.13 for this application of the equipment, it was cited for other applications at this plant. This reference demonstrates the ability of the equipment to function when exposed to a severe steam environment.

This equipment controls the fan cooler SW return valves, opening them fully upon SI initiation. The Licensee states that a review of the manufacturer's drawings indicates that this equipment contains no materials that would be substantially affected by the radiation levels resulting from the DBE. The Licensee further states that the valves (controlled by the transducers) are normally open and that air is removed from both the controllers and the valves following SI initiation. The Licensee concludes that there is no known failure mode to position the valve in the unsafe position. If the NRC concurs in the Licensee's position that subsequent operation of this equipment is not required, the equipment can be considered exempt from qualification requirements.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

The Licensee states that there is no known failure mode for this positioner to place the valve in an unsafe position. FRC agrees with the Licensee's statement (see Appendix D, Item D-10). This equipment item is required for normal plant operation only and is not required to mitigate the consequences of an accident. Therefore, equipment qualification is not required, in accordance with the criteria presented in the DOR Guidelines.

FRC CONCLUSION:

This equipment item is assigned to NRC Category III. FRC agrees with the Licensee's statement that there is no known failure mode which would place this valve in an unsafe position (see Appendix D, Item D-10).

4.4.2 Equipment Item No. 45
Limit Switches Located in the Pipe Penetrations Area
NAMCO Model D2400X
Signals Positions of Containment Ventilation Purge Exhaust Valves (FCV-1171, 1173)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.1.5):

The Licensee states that these limit switches are for position indication only and that the valves within containment served by these items are closed and deactivated on SI and/or containment isolation signal. It further states that there is no known failure that would cause the valve to open. Provided that the NRC agrees that valve position indication is not an essential safety function, FRC finds that qualification is not required for this item.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

These valves perform as containment isolation valves and are closed during power operation. The Licensee has stated that power is administratively removed from the valve actuator and the valves are shut and not used. The implication is that these valves are the same as manually closed containment isolation valves. Since these valves are shut and then de-energized, position indication is not required to verify containment isolation (see Section D.5 of Appendix D).

FRC CONCLUSION:

This equipment is assigned to NRC Category III. FRC concurs with the Licensee's position that containment purge valve position indication need not be environmentally qualified provided the Licensee verifies that appropriate technical specifications and/or procedures preclude opening of these valves during reactor operation.

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

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

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

The qualification of the equipment items in this section has been judged deficient or inadequate based upon a 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 determine the specific qualification category of the equipment item.

For Indian Point Unit No. 3, no equipment falls within this category.

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

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

4.5.2.1 Equipment Item No. 6
Motorized Valve Actuators Located Within Containment
Limitorque Model SMB-00 with Class B Motor Insulation
Actuates RHR Loop Flow Control Valves (HVC-638, 640)
(Licensee References 2.1 and 2.19; Added Final Reference 10.1))

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

Reference 2.1 contains a test report for an   actuator with a
Electric Co. motor with Class B insulation. Reference 2.19 is a

second test report for an SMB-0 actuator with a Reliance motor with Class B insulation. The test sequence of Reference 2.19 is not as severe as that required to simulate the expected environments within the containment. Therefore, the results of Reference 2.19 cannot be used as proof of qualification. With regard to Reference 2.1, FRC has the following comments:

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

b. The referenced test program consisted of high temperature functional tests, a steam exposure including interim functional testing, accelerated aging (stated to be equivalent to 40 years), seismic test, a 150 open/close life-cycle test, and a steam exposure that included boric acid spray.

As a result of this test program, Westinghouse (in Licensee Reference 2.1) stated that actuators with Class B motor insulation will be supplied "where only short term (less than 12 hours) operation is required." The Licensee has stated that the required operating time for these equipment items is less than 8 hours and that it is not normally used during an accident. The NRC should determine the acceptability of the Licensee's position. The Guidelines state that a failure at any time during a test should be considered inconclusive with regard to demonstrating the ability of the component to function for the entire period prior to the failure.

c. Licensee Reference 2.1 also states that another Class B motor (the motor manufacturer was not stated; from the serial number, FRC deduces that this was a motor) was exposed to a radiation dose level of No difference in response was detected between this unit and an identical one that was not irradiated.

However, this similarity of response included overheating, production of smoke, and a transient open circuit during a severe test program involving 220-V operation (instead of 440 V, as in the plant), 45 forward and reverse (f/r) cycles while at room temperature, and ambient. In order to use this test to establish qualification for the installed units, the Licensee should provide an analysis demonstrating that the materials are the same in both the motor installed in the units in the plant and the motor that was irradiated to in the test.

The Licensee has made a commitment to replace the actuators identified as Equipment Item No. 6 with new units having Class RH motor insulation and has cited Licensee Reference 2.2 as evidence of qualification. This reference was discussed in the preceding subsection (3.3.2.1) [4.3.3.1 in this report]. The size (model) of the test specimen is different from the installed equipment. The Licensee should provide an analysis justifying the validity of the cited reference.

LICENSEE RESPONSE:

With respect to the replacement of the motor operators committed, Limitorque Report B0058 [10.1] again addresses the difference in the tested units and units which are to be installed.

FRC EVALUATION:

Reference 10.1 (Limitorque Report B0058) provides a generic discussion of Limitorque's approach to qualification of its equipment. This report states:

"The qualification of the Limitorque Size SMB-0, as reported in the documentation of each of the four tests, was used to generically qualify all sizes of Limitorque operators for the environmental test conditions in accordance with IEEE 382-1972. The Size SMB-0 actuator is an average mid-size unit, and all other sizes of the type SMB, SB, SBD and SMB/HBC are also deemed qualified. All sizes are constructed of the same materials with components designed to equivalent stress levels, same clearances and tolerances with the only difference being in physical size which varies corresponding to the differences in unit rating."

For this equipment, FRC accepts the validity of the approach by Limitorque of qualifying a generic "family" of different size units based upon the tests of the mid-range size.

The Licensee has not assessed aging degradation. The Guidelines require that this be done, considering the long-term exposure of this equipment to temperature, nuclear radiation, and other environmental parameters. From appropriate aging data, the Licensee should establish a conservative qualified life.

As discussed in Section D.4 of Appendix D, FRC does not agree with the Licensee's stated position that this equipment is needed for less than 8 hours. FRC agrees that this equipment should be replaced, as the Licensee proposes to do, since it may be needed for the long term.

FRC CONCLUSION:

The equipment item is not suitable for long-term operation but is assigned to NRC Category IV.b as there is a high likelihood that the short-term function will be performed. The Licensee has committed to replace this equipment with qualified units. The Licensee should establish a conservative qualified life (see Section 4.1.3).

4.5.2.2 Equipment Item Nos. 11A, 11B, 12, and 16B
Electronic Transmitters Located Within Containment
11A & 11B: Foxboro, Type E13DM (MCA)
11A: Pressurizer Level (LT-459, 460, 461)
Steam Generator Level (LT-417A-D, 427A-D, 437A-D, 447A-D)
11B: High Head SI Flow (FT-924A, 925, 926, 926A, 927, 980, 981, 982)
Recirculation Spray Flow (FT-945A, 945B)
12: Foxboro, Type E11GH
RCS Pressure (PT-402, 403)
16B: Foxboro, Type E11GM (MCA)
Pressurizer Pressure (PT-445, 456, 457, 474)
(Licensee References 2.3 and 2.6)

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

The Licensee has not stated the specific monitoring and control functions associated with these transmitters. Typically, these functions would include indication, alarm, reactor trip, safety injection initiation, and post-accident monitoring, with multiple functions sometimes being provided by the same transmitter.

The Licensee has cited References 2.3 and 2.6 as evidence of qualification. Reference 2.3 is not applicable to the Indian Point No. 3 Station because it describes testing results and NRC resolutions concerning

Foxboro Model 613HM transmitters in use at the Indian Point No. 2 Station. Reference 2.3 also states that this model is not used at Indian Point No. 3. Reference 2.6 is a Westinghouse topical report that 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 a half-hour after the accident. FRC's review of this reference has resulted in the following comments:

a. The information base is not easily ascertained. Reference 2.6 contains descriptions of and results from the following qualification programs conducted for the Foxboro Company by various test organizations:

   o Report No. Q9-6005 -- A LOCA exposure test was conducted (excluding radiation and chemical spray) on E13DM, E11GH, and E11GM model transmitters (10-50 mA dc). All units used the standard N0143S4 amplifier.

   o Report No. T3-1013 -- A LOCA exposure test was conducted (excluding radiation) on E13DM, E13DH, E11GH, and E11GM model transmitters (4-20 mA dc). A Conax junction box assembly was also tested. The units utilized amplifier part numbers N0148ND, N0148PF, and N0148NL.

   o Report No. T3-1068 -- A radiation exposure test was conducted on E13DM and E13DH model transmitters (4-20 mA dc and 10-50 mA dc). The units used amplifier part numbers N0148ND, N0148NL, and N0148PD.

   o Report No. T3-1097 -- A radiation exposure test was conducted on improved amplifiers, modified due during the previous test.

   o Report No. T4-6040 -- A dry oven bake, radiation, and hydrostatic test was conducted on E11GM box cover assemblies and associated "E" capsules, O-rings, and seals.

b. Reference 2.6 has presented the results of a variety of tests conducted on Foxboro 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 at the Indian Point No. 3 plant. The Licensee has identified the Foxboro 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 and those of the installed equipment. Therefore, an independent conclusion cannot be reached regarding the extent to which the two units are 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:

- o The full model number for all transmitters (for example, E13DM-1SAM2).
- o The transmitter case style (for example, A or B).
- o The transmitter current output rating (for example, 4 to 20 mA dc or 10 to 50 mA dc).
- o The transmitter top works amplifier part number (for example, N0148PW).
- o The transmitter body material (for example, aluminum, iron, or stainless steel).
- o The transmitter capsule assembly part number and O-ring part number (and material).
- o The method of electrical connection and associated accessories (for example, Conax fitting and pressure seal junction box assembly).
- o 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. The Licensee has stated that these transmitters will not become submerged, and therefore FRC concludes that submergence testing is not required.

e. The Guidelines require that the test chamber temperature/pressure profile envelop the service conditions for a time 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; therefore, 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 transmitters was exposed

   It should be noted that the output error for the

   This is presumably acceptable.

g. Test Report No. T3-1013 states that three Conax connector and junction box assemblies were separately subjected to the same environmental test as the Foxboro transmitters. The Foxboro Co. description of the test states that 3-XJB-I/25 MCA cast iron junction boxes and pressure seal assemblies (including N0148PQ terminal blocks) were tested; however, no reference was made to Conax. The assembly performance was satisfactory. In Reference 2.6, Westinghouse states that Conax 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 Teflon bridge) and employing Teflon and Bishop 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 Foxboro "E"-series 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:

This diode is used in all three amplifier models.

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 that can be degraded by irradiation.

The Licensee has not provided an analysis showing that 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 were incurred during the previous test program (T3-1068),

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 E11GM transmitters having 316 stainless steel cover assemblies. Four "E"-capsule assemblies used silicone elastomer O-rings, part numbers P0120FS and P0120EW; the other four "E"-capsule 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 silicone rubber O-ring, part number U102MV, 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 that:

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 are 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. For the high head and recirculation spray flow transmitters, the Licensee has stated that flows will be established and adjusted following changeover to recirculation and that long-term monitoring of these flows is not required. The NRC will have to judge the validity of this statement and determine the minimum operating time to be required of these transmitters.

7. The Licensee should address the matter of qualified life.

It is FRC's understanding that the Licensee is investigating the need and specific time duration for post-accident, long-term monitoring. In addition, the Licensee has stated that these transmitters will be replaced with sequentially qualified Barton transmitters. The Licensee should provide specific details.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

The Licensee has not provided a response to FRC's DITER. Additional references as evidence of qualification have not been provided. Therefore, the specific deficiencies identified in the DITER remain. The specific areas of deficiency cited were: (i) the exact relationship of the installed transmitter to the appropriate test specimen, including methods of electrical connection and evidence of special MCA modifications and radiation-resistant amplifiers, has not been established; (ii) the test sequences should have included significant irradiation exposure prior to or concurrent with temperature/pressure testing; (iii) transmitter models E11GM and E11GH are deficient with respect to stability; and (iv) aging degradation and qualified life have not been addressed.

It is FRC's judgment that sufficient qualification documentation has been provided to indicate that, because of the comprehensive test program and subsequent results for the Foxboro E series transmitter, there is a high likelihood of short-term operability for these units. The Licensee must provide the necessary qualification information outlined herein in order to adequately resolve the qualification deficiencies.

FRC notes that the Licensee has stated that these transmitters will be replaced with sequentially qualified Barton transmitters; however, a replacement schedule has not been provided.

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b. The qualification for this component is deficient with respect to establishment of an exact relationship between the test specimen and the installed transmitter. In addition, evidence of special MCA modifications and radiation-resistant amplifiers, type of electrical connection, and an adequate test sequence have not been provided. However, this equipment is judged to have a high likelihood of short-term operability because of the comprehensive type test program and results presented for the Foxboro E series transmitters. Aging degradation has not been addressed. The Licensee has stated that these transmitters will be replaced with qualified Barton transmitters; however, a replacement schedule has not been provided. Qualified life of the replacement transmitter should be addressed by the Licensee (see Section 4.1.3).

4.5.2.3 Equipment Item Nos. 14A and 15
Pressure and Flow Transmitters Located in the Auxiliary Pump Room
14A: Foxboro, Type E11GM
Steam Pressure to AFP (PT-1126)
City Water to AFP (PT-1205)
AFP Discharge Pressure (PT-406A&B)
AFP Suction Pressure (PT-1263, 1264, and 1265)
AFP Discharge Pressure (PT-1260, 1261, and 1262)
15: Foxboro, Type E13DM
Auxiliary Feedwater Flow (FT-1200, 1201, 1202, and 1203)
(Licensee reference not cited)

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

The Licensee states that this equipment would not be exposed to a significant increase in ambient temperature during a design basis event when its functioning is required. FRC has not had the opportunity to thoroughly review the plant arrangement and the "systems aspects" of the situation to verify the Licensee's analysis. Based upon a preliminary review, however, FRC has some concerns with regard to the possibility of a significant steam or water jet environment being present in this space as a result of a steamline or feedline break. FRC also notes that Licensee Reference 2.6 applies to this equipment, even though the Licensee did not cite it for these items. Although there will not be an adverse nuclear radiation environment in the auxiliary pump room, FRC believes the Licensee should either (i) provide a stronger justification for the belief that a "harsh" environment cannot exist or (ii) provide an analysis demonstrating that the largest expected shift in the output signal that could occur as a result of a steam or water jet environment will not adversely affect the safety functions of these equipment items.

LICENSEE RESPONSE:

In response to the concern of water and steam jetting, the justification for not being concerned with the jetting in the auxiliary pump room is discussed in the Analysis of High Energy Lines prepared for Con Edison by United Engineers dated May 3, 1973, which has been docketed. Two redundant valves in the main steam supply line to the auxiliary feed pump turbine outside this room have been installed. Each valve is signaled to close automatically on high temperature by its own temperature sensor located in the auxiliary feed pump room. Each valve has control room indication, control, and alarm. Each system is completely independent of the other. Therefore, with the closure of the isolation valves upon a steam line break, steam jetting would be eliminated. Water jetting has not been included within the scope of the DOR Guidelines. Loss of these transmitters will not prevent the ability to safely bring the plant to cold shutdown since manual control capabilities exist.

In addition, it is felt worthwhile to reiterate note "r" of our response, dated April 28, 1980, which states, "Since a concurrent break in the main steam line, feedwater system or a loss of coolant accident in conjunction with a break in the auxiliary feedwater pump steam line results in a double failure and is not considered using single failure criteria, a break in the auxiliary feedwater pump steam line is the accident. With this break, the auxiliary feed system is not initiated because of the small loss of steam, followed by isolation by valves 1310 A and B. Therefore, there is no need for pumps to start and valving to change position."

FRC EVALUATION:

The Licensee Response to FRC's DITER describes a mechanism (redundant valves isolated by a temperature switch actuator) whereby steam jetting and a prolonged harsh environment are eliminated in this area. The Licensee further states that loss of these transmitters will not prevent bringing the plant to cold shutdown since manual control capabilities exist. FRC has analyzed this response and has provided detailed comments in Appendix D, Item D.1. In summary, FRC has concluded that the subject area temperature switches and the redundant valves must be added to the safety-related equipment listing and must be environmentally qualified in order to ensure that the probability of an adverse environment in this area is minimized.

The Licensee has not cited evidence of qualification, but References 2.6 and 2.7 apply to this component. With respect to References 2.6 and 2.7, FRC notes:

- The exact relationship between the installed transmitter and the variety of test specimens has not been established by either Reference 2.6 or 2.7.
- Reference 2.6 has established that the test chamber temperature/pressure profile under all steam conditions exceeded the postulated accident environmental profile for this area. FRC therefore concludes that this aspect of the qualification program is acceptable.
- Qualification for radiation is not required because the transmitter will not be subjected to irradiation.

- Reference 2.6 test results indicated that the E11GM transmitters are
  under
  high-temperature/pressure steam conditions; however, FRC concludes that this deficiency can be dismissed because the transmitters are located in an area where the specified environmental service conditions are not severe for a prolonged period.

From this review, FRC judges that sufficient qualification documentation has been provided to indicate that, because of the comprehensive test program and subsequent results for the Foxboro Series E transmitters, there is a high likelihood of intermediate-term operability for these units.

The Licensee should provide the necessary documentation to clearly establish the exact relationship between the installed transmitter and the test specimen (see Equipment Item Nos. 11A and B) and should address qualified life (see Section 4.1.3).

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b. The qualification for this component is deficient with respect to documentation that clearly establishes the relationship between the installed transmitters and the test specimen. However, this equipment is judged to have a high likelihood of operability for an intermediate (days) duration because the specified environmental service conditions are not severe. A conservative qualified life should be established (see Section 4.1.3).

4.5.2.4 Equipment Item Nos. 14B and 14C
Pressure and Flow Transmitters Located in the Auxiliary Pump Room
14B: Foxboro, Type E11GM
Main Steam Pressure (PT-419 A, B, and C; PT-429 A, B, and C; 449 A, B, and C)
Steam Generator Feedwater Pressure (PT-1163, 1164, 1165, and 1166)
14C: Foxboro, Type E11DM
Main Feedwater Flow (FT-418 A and B, 428 A and B, 438 A and B, 448 A and B)
(Licensee Reference 2.7)

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

Reference 2.7 is a report of a test in which two electronic transmitters were tested by immersion in high-temperature silicone oil. FRC has the following additional comments:

a. The design basis environmental temperature and duration was exceeded by a large margin. This test was probably more severe than a steam exposure test.
This amount of error does not appear to be significant, but acceptance criteria was not given by the Licensee.

b. The test specimens were not identical to the installed equipment. The Guidelines require that the test specimen 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 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 similarity exists, and the validity of the test, as evidence of qualification, has not been established.

c. FRC also notes that Licensee Reference 2.6 applies to this equipment, even though the Licensee did not cite it for these items. Although there will not be an adverse nuclear radiation environment in the auxiliary pump room, FRC believes the Licensee should either (i) provide a stronger justification for the belief that a "harsh" environment cannot exist or (ii) provide an analysis demonstrating that the largest expected shift in the output signal that could occur as a result of a steam or water jet environment will not adversely affect the safety functions of these equipment items.

LICENSEE RESPONSE:

In response to the concern of water and steam jetting, the justification for not being concerned with the jetting in the auxiliary pump room is discussed in the Analysis of High Energy Lines prepared for Con Edison by United Engineers dated May 3, 1973, which has been docketed. Two redundant valves in the main steam supply line to the auxiliary feed pump turbine outside this room have been installed. Each valve is signaled to close automatically on high temperature by its own temperature sensor located in the auxiliary feed pump room. Each valve has control room indication, control, and alarm. Each system is completely independent of the other. Therefore, with the closure of the isolation valves upon a steam line break, steam jetting would be eliminated. Water jetting has not been included within the scope of the DOR guidelines.

FRC EVALUATION:

The Licensee Response to FRC's DITER describes a mechanism (redundant valves isolated by a temperature switch actuator) whereby steam jetting and a prolonged harsh environment are eliminated in this area. FRC has analyzed this response and has provided detailed comments in Appendix D, Item D.2. In summary, FRC has concluded that the subject area temperature switches and the redundant valves must be added to the safety-related equipment listing and must be environmentally qualified in order to ensure limited adverse environmental parameters in this area.

With respect to environmental qualification of the transmitters, both References 2.6 (see Equipment Item Nos. 11A and B) and 2.7 must be used in evaluating this component. Regarding these References, FRC notes that:

- The exact relationship between the installed transmitters and the variety of test specimens has not been established by either Reference 2.6 or 2.7.

- Reference 2.6 has established that the test chamber temperature/pressure profile under all steam conditions exceeded the postulated accident environmental profile for this area. FRC therefore concludes that this aspect of the qualification program is acceptable.

- Reference 2.6 test results indicated that the E11GM and E11GH transmitters are

  under high-temperature/pressure steam conditions; however, FRC concludes that this deficiency can be dismissed because the transmitters are located in an area where the specified environmental service conditions are not severe for a prolonged period.

- Qualification for radiation is not required because the transmitter will not be subject to irradiation.

From this review, FRC judges that sufficient qualification documentation has been provided to indicate that, because of the comprehensive test program and subsequent results for this Foxboro Series E transmitter, there is a high likelihood of intermediate-term operability for these units.

The Licensee should provide the necessary documentation to clearly establish the exact relationship between the installed transmitters and the test specimen (see Equipment Item Nos. 11A and B). In addition, qualified life and aging degradation should be addressed by the Licensee.

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b. The qualification for this component is deficient with respect to documentation that clearly establishes the relationship between the installed transmitters and the test specimen. However, this equipment is judged to have a high likelihood of operability for an intermediate (days) duration because the specified environmental service conditions are not severe. Aging degradation should be addressed and a conservative qualified life established (see Section 4.1.3).

4.5.2.5 Equipment Item No. 13
Pressure Transmitters Located in the Pipe Penetrations Area
Foxboro, Type E11GM
Containment Pressure (PT-948 A, B, and C and 949 A, B, and C)
(Licensee Reference 2.6)

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

Reference 2.6 is a report of several separate tests on Foxboro transmitters, as discussed previously in subsection 3.3.2.3 [4.5.2.2 of this report]. Many of the same concerns listed there apply here also. The location of these transmitters does not experience significant environmentally produced stress during the initial portion of a LOCA (the time during which its most essential function must be performed). In the long term, however, the environment may become severe enough to significantly affect the accuracy of the signal produced. These transmitters provide the containment pressure signal and therefore are of more importance in the long term than many of the other transmitters in the plant. The Guidelines require that the Licensee indicate that ongoing surveillance and maintenance programs are in existence to assure that items exhibiting aging effects are identified and replaced as necessary. The NRC should resolve whether the Licensee should also provide documentation demonstrating the long-term qualification of these transmitters in this location or whether the short-term period of performance stated by the Licensee is sufficient.]

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

The Licensee has not provided a response to FRC's DITER. Additional references as evidence of qualification have not been provided. Therefore, the specific deficiencies identified in the DITER remain. The specific areas of deficiency cited (as stated in Equipment Item Nos. 11A and B) were: (i) the exact relationship of the installed equipment to the appropriate test specimen (including methods of electrical connection, evidence of special MCA modifications, and radiation-resistant amplifiers) has not been established; (ii) the test sequence should have included significant irradiation exposure prior to or concurrent with temperature/pressure testing; (iii) transmitter models E11GM and E11GH are deficient with respect to stability; and (iv) aging degradation and qualified life have not been addressed.

The Licensee has stated in Reference 17:

> The containment pressure is monitored by six Foxboro transmitters Model E11GM. The transmitters initiate a safety injection signal at 3.5 psig and will generate a containment spray signal when the pressure builds up to approximately 22 psig. The transmitters are located outside of containment with a sensor line in containment; therefore, they will not see the adverse containment environment and do not require either short or long term qualification.

Sufficient qualification documentation has been provided to indicate that, because of the comprehensive test program and subsequent results for the Foxboro Series E transmitters, there is a high likelihood of short- to intermediate-term operability for these units.

The Licensee has stated that all environmental parameters, except radiation, remain at normal levels for this area during accident conditions. FRC's comments on this statement are contained in Section 4.1.2. With respect to radiation, the maximum integrated dose level for this area is less than the radiation dose levels for which the test specimen indicated satisfactory performance. Although the comprehensive test results indicate that the transmitters are and therefore the test program should have included significant irradiation exposure prior to or concurrent with the steam-temperature/pressure testing, FRC concludes that the specific test sequence is acceptable because the environmental service conditions are not as severe as the test program.

The Licensee should provide the necessary documentation to clearly establish the exact relationship between the installed transmitters and the test specimen (see Equipment Item Nos. 11A and 11B). In addition, the Licensee should address qualified life and aging degradation.

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b. The qualification for this component is deficient with respect to documentation that clearly establishes the relationship between the installed transmitters and the test specimen. In addition, evidence has not been provided to show that the special MCA modifications and radiation-resistant amplifiers have been incorporated into the design of the installed units. However, this equipment is judged to have a high likelihood of operability for a short-to-intermediate (days) duration because the specified environmental service conditions are not as severe as the test program. Aging degradation has not been addressed. A conservative qualified life should be established (see Section 4.1.3).

4.5.2.6 Equipment Item No. 16A
Pressure Transmitters Located in the Safety Injection Room
Foxboro Type E11GM
Safety Injection Pump Suction and Discharge Pressure (PT-922, 923, and 947)
(Licensee Reference 2.6)

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

Reference 2.6 is a Westinghouse topical report of several separate tests on Foxboro transmitters and was discussed previously in subsection 3.3.2.3 [4.5.2.2 of this report]. Many of the concerns listed there apply for these equipment applications also. The Licensee states that these transmitters are required for 30 days (i.e., long term). Although nuclear radiation levels will not be as high as they are within containment following a LOCA, the SI Pump Room will experience a combination of significant radiation dose rates, elevated temperatures, and high humidity. These environments should be qualified by the Licensee (note also a concern regarding nuclear radiation dose levels expressed in Section 4), and qualification should be demonstrated.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

The Licensee has not provided a response to FRC's DITER. Additional references as evidence of qualification have not been provided. Therefore, the specific deficiencies identified in the DITER remain. The specific areas of deficiency cited (as stated in Equipment Item Nos. 11A and 11B) were: (i) the exact relationship of the installed equipment to the appropriate test specimen (including methods of electrical connection, evidence of special MCA modifications, and radiation-resistant amplifiers) has not been established; (ii) the test sequence should have included significant irradiation exposure prior to or concurrent with temperature/pressure testing; (iii) transmitter Models are ; and (iv) aging degradation and qualified life have not been addressed.

Sufficient qualification documentation has been provided to indicate that, because of the comprehensive test program and subsequent results for the Foxboro Series E transmitter, there is a high likelihood of short- to intermediate-term operability for these units.

The Licensee has stated that all environmental parameters, except radiation, remain at normal levels for this area during accident conditions. With respect to radiation, the maximum integrated dose level for this area is less than the radiation dose levels for which the test specimen indicated satisfactory performance. Although the comprehensive test results indicate that the transmitters are susceptible to degradation by radiation exposure and therefore the test program should have included significant irradiation exposure prior to or concurrent with the steam-temperature/pressure testing, FRC finds the specific test sequence acceptable because the environmental service conditions are not as severe as the test program.

The Licensee should provide the necessary documentation to clearly establish the exact relationship between the installed transmitters and the test specimen (see Equipment Item Nos. 11A and 11B). In addition, the Licensee should address qualified life and aging degradation.

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b. The qualification for this component is deficient with respect to documentation that clearly establishes the relationship between the installed transmitters and the test specimen. In addition, evidence has not been provided to show that the special MCA modifications and radiation-resistant amplifiers have been incorporated into the design of the installed units. However, this equipment is judged to have a high likelihood of operability for a short-to-intermediate (days) duration because the specified environmental service conditions are not severe. Aging degradation should be addressed and a conservative qualified life established (see Section 4.1.3).

4.5.2.7 Equipment Item Nos. 21, 22, 24, and 25
Solenoid Valves Located in the Pipe Penetrations Area
21: ASCO Model 8314
22: ASCO Model 8316
24: ASCO Model 8317
25: ASCO Model 8300
Actuate 49 Separate Isolation Valves
(Licensee References 2.4 and 2.5)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.10):

Reference 2.4 is an analysis of the solenoid valves' ability to perform the safety function of de-energizing and venting the diaphragm of an air-operated valve. Reference 2.5 is a statement by the Licensee concerning the sensitivity of valve materials to irradiation. FRC's comments are as follows:

a. In Reference 2.4, it is noted that the force used in performing the solenoid valve safety functions is provided by a spring. This analysis gives some confidence that this equipment will perform its safety function without the occurrence of a common-mode failure if the required function occurs before the environment becomes significantly degraded. However, the Licensee states that the operation of this equipment is needed over the long term following a LOCA.

b. In Reference 2.5, the Licensee claims radiation sensitivity levels of 6×10^6 rd for Buna-N and 5×10^7 rd for nylon (the materials identified as being present in the equipment). These levels are greater than the predicted exposures at this location in the plant. Appendix C of the DOR Guidelines indicates that these materials have threshold radiation susceptibility levels lower than the long-term integrated dose levels at this location and that these materials are susceptible to thermal aging degradation. In addition, IE Bulletin 78-14 notes a number of failures of solenoid valves containing Buna-N as a result of its aging characteristics.

c. Aging degradation has not been considered; qualified life has not been established, nor is there 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.

LICENSEE RESPONSE:

This paragraph implies that the degraded environment will affect the metallic spring. We do not see the reason for this conclusion since the environment will not have an adverse effect on a metallic spring.

FRC EVALUATION:

There is concern that the radiation environment may affect the seals and lubricants, causing the plunger to bind even though a spring force is applied. Also, many of these valves may have to be reopened (e.g., to obtain samples) at some time subsequent to their closure (the Licensee does not explicitly address this but does state that they are needed for "30 days," i.e., long term). For this reason, FRC believes that it is necessary for the Licensee to establish environmental qualification of this equipment for the conditions to which the valve is subject on a more rigorous basis than has been done thus far. Aging degradation and qualified life should be evaluated by the Licensee. (It should be noted that the manufacturer can provide information on qualified life and replacement schedules.)

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b. Although qualification documentation has not been provided, FRC believes it is likely that these solenoid valves will function satisfactorily because the environment is nonharsh except for radiation. The Licensee should establish a qualified life of non-metallic parts, based on manufacturer data, and institute a replacement schedule, if needed.

4.5.2.8 Equipment Item Nos. 35 and 36
Electric Motors Located Within Containment
35: Westinghouse, 588-5 Frame
SI Recirculation Pump Drive
36: Westinghouse, 69F97009
Fan Cooler Motor
(Licensee References 2.11 and 2.12)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.17):

FRC's review of the referenced documents has resulted in the following observations:

a. Reference 2.11 states that its purpose was to support qualification of reactor containment fan coolers, ice condenser air recirculation motors, and recirculation and spray pump motors. The Licensee has neither established that any of the units tested are the same as these equipment items nor provided an analysis to show a relationship between them.

   The test program described in this reference was quite comprehensive and included pre-aging for a 40-year installed life. The LOCA simulation exceeded the plant requirements by a wide margin. The test program is considered fully satisfactory for the motor subjected to the test.

b. Reference 2.12 is a report of a test program intended to establish the effect of irradiation on the insulation system. It provides additional confidence that this type of insulation system should perform as expected under accident conditions. Again, there is insufficient information to determine the applicability of this reference to the installed equipment.

LICENSEE RESPONSE:

We are obtaining further documentation from Westinghouse to substantiate the conclusions that the motors are qualified under the documentation in our submittal dated April 28, 1980.

FRC EVALUATION:

FRC has reviewed the June 13, 1979 letter [6] from PASNY to the NRC which stated that the safety-injection recirculation pump motor and the containment recirculation fan motor were the basic generic type and were qualified by Westinghouse Report WCAP-7829 [2.11]. PASNY referenced evidence of the applicability of these tests to the plant's motors through additional letters from Westinghouse which were previously sent to the NRC (Westinghouse Reference letters NS-CE-1009 dated March 31, 1976; N-CE-728 dated August 1, 1975; and NS-CE-692 dated July 10, 1975). A review by FRC of these letters fails to indicate that the test specimens were the same as the motors in the plant. The Licensee, in a follow-up telephone conversation, indicated that investigation relative to the test specimens would continue.

The Licensee described the motors as furnished with an integral mounted water cooler [9]. This information is important because it determines whether long-term availability can be demonstrated. Details identifying the type of lubricant, bearings, motor-lead splices, and lead-to-cable splices were also lacking. These details must be provided before qualification can be considered adequate.

The Licensee has not addressed the DOR Guidelines requirement that a definitive qualified life statement be provided, as outlined in Section 4.1.3 of this TER.

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b because the motor manufacturer has conducted testing on similar motors, but traceability for these motors has not been demonstrated. A qualified life statement has not been provided and aging mechanisms have not been addressed.

4.5.2.9 Equipment Item Nos. 40A and 40B
Electrical Cables and Splices Located Within Containment
40A: Cable Manufacturer Not Stated/Raychem Splices
40B: Kerite Company 600 V, Multiconductor Power and Control Cable, #12 AWG/Raychem Splices
(Licensee References 2.1, 2.22, and 2.23)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.21):

Reference 2.22 defines procedures to be used for making splices and is not directly relevant to qualification. FRC's comments on References 2.1 and 2.23 follow:

a. Licensee Reference 2.1 is a Westinghouse Topical Report that describes a test program in which samples of No. 12 AWG 2/c cables, plus others (No. 4 AWG 1/c and No. 12 AWG 1/c, silicone, no manufacturer stated), taken from the Indian Point Unit 2 plant site during construction, were spliced by Raychem and then exposed to various combinations of thermal aging (stated to be the equivalent of 40 years of actual service), gamma irradiation, and steam/chemical spray. The temperature/pressure parameters envelop those specified by the Licensee. The chemical spray composition in the test was slightly different from that at the plant, the spray duration was somewhat shorter (2 hours vs. 2.5 for the installed equipment), and the spray density was not stated. These deviations are judged to be acceptable (other tests in the same series, reported in the same reference, did use the proper spray density).

The cable samples were electrically loaded during the steam/chemical spray exposure by having 480 V applied to them. The No. 12 AWG samples had a current of 13 to 15 A passed through the conductors, while the No. 4 AWG conductors carried approximately 50 A.

, the electrical loading was applied only during the first 2 hours of the exposure, and 8 hours per day during the final 16 days of the 3-week total exposure period.

While the interruption in the application of simulated operational service conditions is an undesirable occurrence, it is not regarded as a serious deviation from the Guidelines. The Licensee has not provided information concerning the actual operational service conditions for the installed cables (specifically, the current load). Thus, FRC cannot judge the adequacy of the current load applied during the test in producing insulation overheating representative of actual conditions, especially where power cables are installed in sealed (or nearly sealed) conduits.

The nuclear radiation exposure of the cable samples was 50 Mrd, using a gamma radiation source, and the other samples received 200 Mrd. The Licensee has not made a specific evaluation of the contribution to the total dose experienced by the cable that results from beta radiation, as is required by the Guidelines. Also, the Licensee has not stated the dose that the cable will accumulate during its installed life under normal plant operation. FRC therefore has no basis for determining whether the nuclear radiation exposures in the test program for the cable samples are adequate. (FRC believes that the dose used for the other cable samples is adequate.)

The Guidelines require that the test specimen must be the same as the equipment being qualified. The tested cable sample was 2 conductor, while the Licensee's submittal refers to "multiconductor cable" for this item. If the Licensee is citing this test as evidence of qualification for other cable constructions (number of conductors, jacket and insulation thickness), an analysis should be provided demonstrating whether differences have any impact on performance under LOCA conditions. Also, uses different formulations in its insulation and jacket for different cable orders. The Licensee has not established that all the Class 1E cables used in Unit 3 are identical to the cable sample taken from Unit 2 that was tested. In the case of the other cables, the Licensee has not established any relationship between some poorly identified cable samples taken from Unit 2 and tested, and all of the other Class 1E power and control cables installed in Unit 3.

b. Licensee Reference 2.23 is a Raychem test report for a comprehensive qualification program. Along with this reference, the Licensee has provided copies of correspondence concerning splices at the plant. One of these indicated that the splices covered in this reference were not used at Unit 3. The packet of material includes letters from Raychem that refer to the splices made on the cable samples used in the test program described in Reference 2.1. If all of the splices in the plant are identical to those tested in either of the cited references, the qualification requirements can be considered to have been satisfied (except for an ongoing program to monitor failures). Unfortunately, the Licensee has not established that this is the case, since a qualified splice involves both the splice materials and the jackets of the cables that are being spliced. As mentioned above, the Licensee needs to make a careful review of all cables connected to Class 1E equipment, tabulate data that describes them (including manufacturer; conductor, insulation and jacket materials; dimensions; etc.), and determine which are spliced. This information and all relevant qualification reports should be provided for review.

c. Qualification for submergence has not been addressed.

LICENSEE RESPONSE:

The TER states that our Reference 2.22 is not directly relevant to qualification. We would like to point out that this document forms the bridge between the type of cables that were tested and installed. The document also spells out splicing procedures which were used in the field; therefore, it is also valuable in order to compare with the tested splice. The document also identifies the manufacturers who supplied the cable by tabulating the associated purchase orders. The following is a comparison between the environment qualification tests performed at Franklin Institute Research Lab and the cables used at IP for safety-related systems.

We concur that Reference 2.23 of our response is not applicable to IP3; this was an oversight.

The Kerite cable received of gamma dose, which is more than adequate to encompass the accumulation of the exposure, stated in the guidelines, of and any additional dosage due to Beta. The installed life dosage is considered in the guidelines-recommended exposure; therefore, further defining this value is pointless.

FRC EVALUATION:

FRC has reviewed the Licensee Response and the conduit and cable schedule for Indian Point Unit No. 3 [2.22], and has reanalyzed the comments of the DITER which are contained in preceding paragraphs. FRC has the following comments:

a. The Licensee submittals have not stated whether the cables within containment are in exposed cable trays, covered cable trays, or conduit. This is a significant factor affecting the radiation exposure and, therefore, the ability of the cables to successfully perform. For equipment that can be exposed to beta radiation, the Guidelines require testing to a dose of at least 160 up to 200 Mrd as opposed to the 20 Mrd or 50 Mrd discussed in the Licensee Response. It is FRC's understanding that the cable trays at Indian Point Unit No. 3 are not covered, and therefore the cables would be exposed to the beta plus gamma dosage.

b. The significance of the beta radiation dosage and dose rate is based on FRC's EEQ review of several test reports from the manufacturers of the cables identified as being inside containment. (see Appendix G).

(1) Regarding Kerite cables stated by the Licensee to be in Indian Point Unit 3, FRC has the following comments.

FRC has reviewed Reference 2.1 and FIRL Reports F-C4158, F-C4020-1, and F-C4020-2 (tests on Kerite cable). The cables covered by these reports are:

F-C4518: 7/C No. 12 [ ]

F-C4020-1: 7/C No. 12 [ ]

F-C4020-2: 7/C No. 12 [ ]

F-C4020-2: 1/C No. 6 [ ]

For the cables reported in FIRL Report F-C4020-1, there was a noticeable decrease in insulation resistance after thermal aging and approximately a factor of 100 reduction in insulation resistance after radiation exposure. The report identifies a further reduction (approximately a factor of) in insulation resistance after the first 1.5-hour exposure to 346°F/113 psig in the test chamber.

Thermally aged 7-conductor cables subjected to the LOCA conditions and irradiation did not pass high potential tests described in FIRL Reports F-C4020-2 and F-C4158 and also showed during the test. Three of the four single-conductor cables tested and reported in FIRL Report F-C4020-2 performed satisfactorily for aged and unaged cables subjected to simultaneous radiation and LOCA.

(2) BIW Report 910 was submitted as evidence of qualification of type cables. This report referenced FIRL Report F-C3859-1 for aging/LOCA/irradiation testing. FRC has reviewed this test report and notes that 17 cables were tested. All cables were aged for a minimum of hours at ; three cables were subjected to an additional hours at . The cables consisted of the following insulation/jacket combinations:

a. insulation/ jacket (1 sample)
b. EPR insulation/ jacket (5 samples)
c. XLPE insulation F.R./ jacket (2 samples)
d. XLPE insulation F.R./Neoprene jacket (4 samples)
e. ETFE insulation/ jacket (1 sample)
f. polyimide insulation/ polyimide jacket (1 sample)
g. insulation.
h. XLPE insulation F.R.
i. , ETFE insulation.

Cables a and f above showed from the circuit during the thermal/radiation aging part of the test after Mrd and thermal aging (cumulative) of

The cables were These data indicate that there is of the type cable as a result of aging and irradiation followed by IEEE Std 323-74 type LOCA conditions.

From the review of cable and electrical equipment testing, FRC is aware of anomalies with some (formulation unidentified) BIW cables under LOCA conditions of approximately psig, spray Two types (2 samples each) showed adequate insulation resistance; The report did not state a reason for the short circuit event.

(3) For Raychem Flamtrol cables, FRC has reviewed FIRL Reports F-C4033-1, F-C4033-2, and F-C4033-3. These tests involved 1/C No. 12 AWG, 1/C AWG No. 6, 7/C AWG 12, 2/C AWG 16, 22 AWG coaxial cable, and 26 AWG triaxial cables. These cables also showed a when subjected to radiation, chemical spray and pressure, temperature, and humidity due to simulated LOCA conditions. The was not as large as described in (1) and (2) above, but it varied with the specimens tested.

FRC CONCLUSION:

This equipment is assigned to NRC Category IV.b because the testing is extensive and establishes a high probability of operability before the radiation dose rate becomes high. FRC considers that this cable could be upgraded to NRC Category II if protection against beta radiation were provided. A qualified life should be established (see Section 4.1.3).

4.5.2.10 Equipment Item No. 41
Electrical Instrument Cables and Splices Located Within Containment
Cable Manufacturer Not Stated/Raychem Splices
(Licensee Reference 2.24)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.22):

Reference 2.24 is a test report for two types of silicone-rubber insulated 2/c cables. The manufacturer, formulation, and details of construction are not stated, but the manufacturer is presumed to be Lewis Engineering Company, since that organization provided the cable samples tested. FRC also notes the following:

a. The cable samples received of gamma radiation prior to the thermal/steam exposure at

Following this, the samples were exposed to A 1-kV potential was applied during the steam and temperature exposures. No failures occurred.

b. The Guidelines state that the test specimen should be the same as the equipment installed in the plant. The descriptions of both the installed equipment and the test specimens are somewhat vague, but seem to refer to different constructions. The Licensee did not present an analysis comparing the impact of deviations between the test specimen's configurations, design features, and production procedures and those of the installed equipment. Hence the validity of the test, as evidence of qualification of the installed cable, has not been established.

c. The qualification program neither addressed thermal aging nor established the qualified life, as the Guidelines require. Also, no electrical loading was applied during the simulated LOCA exposure, and the Licensee neither stated the radiation dose the cable will accumulate during normal operation over its installed life, nor specifically evaluated the beta radiation dose, as required by the Guidelines.

d. No information has been provided concerning the splices on these cables.

e. No information has been provided concerning qualification for submergence conditions.

LICENSEE RESPONSE:

The cable order specification for IP2 and IP3 are identical; therefore, the actual cables must be fabricated with same formulation of insulation and jacket material. With respect to the Lewis Cable Section 3.3.2.22, paragraph b, the statement does not follow. "The description of both the installed cable and test specimen are somewhat vague, but seems to refer to different construction."

FRC EVALUATION:

FRC has reviewed the Licensee Response and the conduit and cable schedule transmitted by the Licensee on December 12, 1980, which resolves the traceability question. FRC has also reevaluated Reference 2.24 which concerns the Licensee statement about possible submergence of cables contained in PASNY letter IP-JCS-9105 dated July 30, 1980. FRC has the following additional comments regarding this equipment item:

a. As noted under Items 40A and 40B, FRC understands that the cable is installed in uncovered cable trays. Therefore, an integrated radiation dose of could be expected as a result of a LOCA. Reference 2.24 documents when the cables were exposed to steam pressure and temperature combined with chemical spray. The tested cables had previously been exposed to gamma radiation.

b. Cable performance has not been established for the reduced insulation resistance condition during a LOCA. Particularly for instrumentation and control applications, the effects of attenuation and distortion of signals should be evaluated (see Appendix G).

c. The effects of submergence during and after a LOCA have not been evaluated, nor does the Reference 2.24 test address the possibility of submergence. FRC is not aware of any testing of submerged cables that would be applicable to this item.

d. None of the testing simulated the installation stresses that would be experienced by the installed cables. The cables in the tests were precoiled by the manufacturer and carefully laid on a simulated cable tray in the test chamber.

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b because the testing is extensive and establishes a high probability of operability before the radiation dose rate becomes high. Satisfactory long-term performance with reduced insulation resistance during LOCA simulation has not been demonstrated. FRC believes that this item could be upgraded to NRC Category II if protection against beta radiation is provided.

4.5.2.11 Equipment Item No. 43
Resistance Temperature Detector Elements
Sostman Type 11901B
(Licensee Reference 2.18)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.25):

Reference 2.18 is Westinghouse Topical Report WCAP-9157 that contains a test report for resistance temperature detectors. There are some unresolved issues concerning this reference that are discussed in an internal NRC memorandum from G. Lainas to D. G. Eisenhut. The primary issue concerns the method of calculating the 40-year normal radiation dose and accident radiation dose for those sensors which are mounted in the primary coolant piping.

The qualification test conditions satisfy the requirements provided by the Licensee for the containment environment, but the maximum temperatures within the primary system piping that must be both survived and sensed have not been identified.

The NRC position on (a) the post-LOCA time duration that the equipment is required to be functional and (b) the environments within the primary system piping must be established before a determination of the qualification of this equipment can be made. Because this equipment contains materials that are subject to aging degradation, these must be addressed and the period of qualified life (and hence replacement schedule) established.

LICENSEE RESPONSE:

This wide range resistance temperature detector elements are on a schedule and are changed out accordingly.

FRC EVALUATION:

The Licensee Response does not answer the questions raised in FRC's DITER, nor have additional references as evidence of qualification been provided. Therefore, the specific deficiencies identified in the DITER remain. Section D.9 in Appendix D details FRC's position regarding the need for this equipment to be qualified for long-term operation.

4

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b. From a review of all documentation, FRC concludes that the qualification of this equipment item for the environmental service conditions at the installed location has not been demonstrated, but it is likely that short-term performance will be satisfactory based on results of the test referenced by the Licensee.

4.5.2.12 Equipment Item No. 23B
Solenoid Valves Located in the Steam and Feedline Penetrations Area
ASCO Model 8300
Actuates Main Feedwater Regulator Valves (PCV-417, 427, 437, 447)
(Licensee Reference 2.4)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.11):

Licensee Reference 2.4 was discussed in the preceding subsection, and this item is the same as Equipment Item No. 25. The Licensee also notes that (1) these valves will fail to the "closed" (safe) position for all potential modes of failures; (2) the ambient environmental conditions in the locations where the valves are installed will not change significantly from the conditions existing during normal operation, when accidents occur; and (3) some of the valves are needed for only a period of a few minutes, while others are needed for a long time following an accident.

FRC has not had the opportunity to thoroughly review the plant arrangement and the "systems aspects" of the situation to verify the Licensee's analysis. Based upon a preliminary review, however, FRC has some concerns with regard to the possibility of a significant steam or water jet environment being present in these locations as a result of a steamline or feedline break.

In addition, as noted above, the following Guidelines requirements have not been addressed: aging degradation has not been considered; qualified life has not been established; and a program has not been established to ascertain whether any in-service failures during the installed life to the equipment are the results of aging degradation. These considerations are of particular concern for solenoid valves that are energized during normal plant operation.

LICENSEE RESPONSE:

This concern is covered by the jet and water discussion in Section 3.3.2.5 [4.5.2.4 of this report].

FRC EVALUATION:

As discussed in Section D.3 of Appendix D, FRC believes that qualification is required for this equipment. As noted in Section 4.1.2, FRC believes the environment under HELB conditions could be "harsh," and not "mild" as the Licensee has stated. No evidence of qualification has been provided. However, the cited reference does provide some assurance that the equipment will function. The Guidelines requirements regarding aging degradation and qualified life should be addressed by the Licensee. The concerns expressed in Section 4.1.3 on these subjects are of particular importance for solenoid valves that are energized during normal plant operation. It is noted that the Licensee Response refers to steam getting in the auxiliary pump room, and the arguments are not valid for the steam/feedline area.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because qualification documentation has not been provided. The Licensee should determine the qualified life of non-metallic parts based on manufacturer's recommendations so that proper maintenance can be scheduled and performed.

4.6 NRC Category V EQUIPMENT THAT IS UNQUALIFIED

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

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

4.6.1 Equipment Item No. 10
Level Switches Located Within Containment
GEMS Corporation Model LS 1900
(Licensee Reference 2.18)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.3.1):

Licensee Reference 2.18 is a Westinghouse Topical Report. FRC has reviewed this document and found no discussion of test data or analysis for any level switches.

LICENSEE RESPONSE:

> NUREG 0578 required installation of an environmentally qualified level system. We are proceeding with the installation of that system. The transmitters are ordered and will be installed at the next outage of sufficient duration.

FRC EVALUATION:

The Licensee has not provided any valid qualification references for the equipment presently installed in the plant.

In the original submittal [1], the Licensee also noted that these instruments were designed for submerged service at 295°F/60 psig. No substantiation has been provided for this claim. The Licensee has cited systems-related reasoning in lieu of qualification documentation. This is discussed in Section D.11 of Appendix D.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V. From a review of all documentation, FRC concludes that this equipment item is not environmentally qualified for the service conditions at the installed location. FRC notes that the Licensee has stated that this item will be replaced with qualified equipment.

4.6.2 Equipment Item No. 17
Flow Transmitters Located Within Containment
Barton Type 386
RHR Recirculation Flow FT-946 A-D
(Licensee References 2.3 and 2.1)

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

Reference 2.3 is an NRC memorandum that discusses actions to be taken concerning certain unqualified electronic transmitters in Indian Point Unit 2. It also indicates that the problem it discusses does not apply to Unit 3. Reference 2.1 is a Westinghouse Topical Report in which a qualification test program for a transmitter is described. The Licensee also has stated that this item is to be replaced with qualified Barton transmitters. FRC has reviewed Reference 2.1 and has the following comments:

a. The test specimen is a different type than the installed equipment. 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. Therefore, an independent conclusion cannot be reached regarding the extent to which the two units are similar and the validity of the test as evidence of qualification has not been established.

b. The temperature/pressure profile

c. There was no chemical spray in the test program.

d. The test program included seismic test and a nuclear radiation dose of 240 Mrd administered after the steam exposure.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

Westinghouse Report WCAP-7410-L documents a test program conducted on under simulated LOCA conditions; however, the time was limited Pre- and post-test accuracies showed FRC previously concluded that it would be advisable for the Licensee to establish the relationship between the installed Barton transmitters and the tested transmitter.

The Licensee has not cited additional references as evidence of qualification for this transmitter in response to FRC's DITER. Neither has the Licensee provided information that would establish similarity between the Barton Model 332 MOD I, which was tested, and the installed Barton transmitter. FRC therefore finds that qualification has not been established for this equipment.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because evidence of qualification has not been made available. The Licensee has stated that this transmitter will be replaced with qualified Barton transmitters.

4.6.3 Equipment Item No. 18B
Solenoid Valves Located Within Containment
Automatic Switch Co. (ASCO) Model NP-8316
Actuates Containment Purge Valves (FVC-1170, 1172) and
Containment Pressure Relief Valve (PCV-1190)
(Licensee Reference 2.8)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.9):

The Licensee's reference is a proprietary test report describing a qualification program conducted for a number of ASCO valves. FRC's review of this report has resulted in the following conclusions:

a. Of the valve models tested, the one with a model number that most closely matches that of the installed equipment is sample No. 6,

solenoid enclosure, and normally closed operation. The Guidelines require that the test specimen be the same as the equipment being qualified. The Licensee did not present information describing the installed item; a statement that it is identical to the test sample; or an analysis comparing the impact of deviations between the test specimen's specific design features, materials, and production procedures and those of the installed equipment. Therefore, an independent conclusion cannot be reached regarding the extent to which the results of the test program provide valid evidence of qualification. The Licensee should provide certification that the important features of the installed equipment are the same as those in the test specimen.

b. The environmental and operational service condition parameters used in the qualification test program exceeded those dictated by plant-specific requirements in all cases except the of the steam temperature/pressure profile. This deficiency is not regarded as being significant. The Licensee submittal did not consider the nuclear radiation dose resulting from (i) normal plant operations and (ii) beta radiation (including the Bremsstrahlung radiation it creates while being attenuated). However, the test program included a sufficiently large gamma radiation dose (Mrd) that the other dose contributions can be considered to have been accommodated.

c. The pre-aging simulated in the test program was intended to represent an installed life (and hence a qualified life) of ambient temperature. The ambient temperatures at the installed locations within the plant are lower, and hence the qualified life is longer. The Licensee should make an explicit determination of the qualified life and establish a replacement schedule if this is less than the period for which the plant is licensed to operate.

LICENSEE RESPONSE:

The concern is that the ASCO solenoid valve NP 8316A75E, which was installed, differs from the tested model NP 831665E. The difference between the valves is the size of the pipe connection and the orifice. The concern of aging is on-going; however, since we have data to indicate that the solenoid will perform its function for a minimum of 4 years, a small replacement schedule is incorporated. This schedule will be modified as necessary when more data on aging is received.

FRC EVALUATION:

The cited reference is valid for this equipment item, because the Licensee has more fully identified the equipment model number. However, the following concerns still remain:

1. During the referenced qualification test, there was This was evidently the result of of conduit material and the method of electrical connection used in the test program, which does not appear to represent that used in any power plant. There is the strong implication that the test was to be conducted with the electrical wiring penetration of the solenoid case isolated (sealed) from the test environment. It was this isolation barrier that evidently failed during the test, allowing spray solution to enter and seriously degrade the coil. Although this did not occur with sample No. 6, which is the one that most closely matches the installed equipment, there is nothing in the referenced report to indicate that this was not merely a fortuitous result. The results of the test must therefore be regarded as inconclusive until the uncertainties associated with the method of making the wiring interface with the solenoid, both in the plant and in the test, are resolved. The Guidelines state (Section 5.2.5):

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

   They further state (Section 5.2.6):

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

2. FRC interprets the Licensee Response to indicate agreement that at present the qualified life is 4 years. The Guidelines require an explicit statement with regard to qualified life.

3. The Licensee did not indicate if subsequent operation of the containment pressure relief valve would be required following an accident.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because adequate evidence of qualification was not provided, including the method of sealing. The Licensee should provide evidence that a qualified electrical connection seal has been used.

4.6.4 Equipment Item No. 29
Position Switches Located in the Auxiliary Pump Room
NAMCO Model EA-170
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.14):

The Licensee claims that, following an accident, the ambient environmental conditions will not be significantly different from those existing during normal plant operation. FRC has not had the opportunity to review the plant arrangement, but is concerned that the existence of a significant steam or water spray condition is possible. Also, the Licensee has not addressed the requirements of the Guidelines concerning aging and has not established the qualified life for this equipment.

LICENSEE RESPONSE:

In response to the concern of water and steam jetting, the justification for not being concerned with the jetting in the auxiliary pump room is discussed in the Analysis of High Energy Lines prepared for Con Edison by United Engineers dated May 3, 1973, which has been docketed. Two redundant valves in the main steam supply line to the auxiliary feed pump turbine outside this room have been installed. Each valve is signaled to close automatically on high temperature by its own temperature sensor located in the auxiliary feed pump room. Each valve has control room indication, control, and alarm. Each system is completely independent of the other. Therefore, with the closure of the isolation valves upon a steam line break, steam jetting would be eliminated. Water jetting has not been included within the scope of the DOR Guidelines.

FRC EVALUATION:

The comments contained in Section D.1 of Appendix D apply to this item in that the equipment used to preclude a harsh environment is not qualified. There is no other information available to FRC regarding the conditions for which the limit switch would be qualified.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V. From a review of all documentation, FRC concludes that this equipment item is not qualified for the environmental service conditions at the installed location.

4.6.5 Equipment Item No. 34A
Large Electric Motor Located in the Auxiliary Pump Room
Westinghouse, 509 US Frame
AFW Pump Drive
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.15):

The Licensee states that the ambient environmental conditions in this location will not change significantly from the conditions existing during normal operation, when accidents occur.

FRC has not had the opportunity to thoroughly review the plant arrangement and the "systems aspects" of the situation to verify the Licensee's analysis. Based upon a preliminary review, however, FRC has some concerns with regard to the possibility of a significant steam or water jet environment being present in this location as a result of a steamline or feedline break and also with regard to the required time-for-functioning cited by the Licensee.

In addition, the following Guidelines requirements have not been addressed: aging degradation has not been considered; qualified life has not been established; and a program has not been established to ascertain whether any in-service failures during the installed life of the equipment are the result of aging degradation.

LICENSEE RESPONSE:

> We are obtaining further documentation from Westinghouse to substantiate the conclusions that the motors are qualified under the documentation in our submittal dated April 28, 1980.

FRC EVALUATION:

References as evidence of qualification have not been provided. Therefore, the specific deficiencies identified in the DITER remain.

The Licensee has requested qualification documentation from the motor manufacturer. The Licensee should address the following related items:

1. provide documentation that a lubricant qualified for the steam environment was used.

2. review maintenance records to determine if abnormal motor component wear conditions are being experienced that could decrease the motor's qualified life (such items as bearings and splices should be specifically addressed)

3. provide motor nameplate information to identify the specific type of motor insulation that was used.

4. provide evidence that the water spray would not affect the operation of the motor.

A review by FRC indicates that Westinghouse Report WCAP-8754, entitled "Environmental Qualification of Class 1E Motors for Nuclear Out-of-Containment Use" may have some applicability to this motor if the Indian Point Unit No. 3 motor uses Class B or LF insulation and has operating speeds ranging from 720 to 3600 rpm.

The Licensee has limited the environmental temperature and pressure in this area by employing a steam isolation valve actuated by a thermostat for which qualification has not been demonstrated. However, the Licensee did not list the thermostat as a safety-related device. The exact environmental service conditions are therefore a concern because operability of the steamline isolation system has not been demonstrated, as is discussed in Section D.1 of Appendix D.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V. The Licensee has not provided qualification documentation and is relying upon an unqualified system to achieve a nonharsh environment. Sections 4.1.2 and D.1 of Appendix D provide additional information.

4.6.6 Equipment Item No. 38
Terminal Blocks Located Within Containment
Westinghouse Model 542247 (805432)
(Licensee References 2.14, 2.15, and 2.16)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.19):

Reference 2.14 is an internal NRC memo dated February 3, 1979 concerning qualification of safety-related terminal blocks for service inside containment at Yankee Rowe Nuclear Power Station. This reference indicates that these blocks have successfully passed a steam exposure test performed in connection with the Haddam Neck plant and that Westinghouse has stated that this item has a total integrated radiation dose capability of This reference cannot be considered to be valid evidence of qualification since it is neither a test report nor an analysis that references test data.

Reference 2.15 is an analysis conducted for Westinghouse terminal blocks installed at the R. E. Ginna station, again based on the tests conducted for the Haddam Neck plant and the Westinghouse statement concerning radiation capability.

Reference 2.16 is the report for the tests performed for the Haddam Neck plant referred to above. The test included only a steam exposure.

FRC's assessment of the status of qualification documentation follows:

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

b. 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. In reviewing qualification data referenced in connection with the Palisades plant, FRC noted that the Westinghouse statement regarding radiation qualification was quoted out of context, and the situation is unsatisfactory for the long term following a LOCA.

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

d. During one of the steam exposure tests in the program described in Reference 2.16, 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. It is noted that use of a resilient washer under the screw head will eliminate this particular qualification-related failure mechanism.

LICENSEE RESPONSE:

All the terminal blocks inside containment are mounted in junction boxes. Resilient washers will be installed under the blocks to preclude any failures.

FRC EVALUATION:

The Licensee has not provided any information in response to the questions raised in the DITER concerning thermal and radiation aging, qualified life, and combined radiation/steam/chemical spray conditions. The Licensee Response addressed the question of installation stresses raised in paragraph 3.3.2.19 but has not addressed the problems of thermal stress cracking such as occurred on some terminal blocks discussed in Reference 2.16.

FRC has the following additional comments on the terminal blocks based on information which has been reviewed in the EEQ program.

1. 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 instrument signals.

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

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

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

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

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

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

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

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because evidence of qualification for Indian Point Unit No. 3 has not been provided. There is no assurance that the terminal blocks will transmit reliable instrument signals under LOCA or HELB conditions.

4.6.7 Equipment Item No. 39
Electrical Penetrations Located Within and Outside Containment
Crouse-Hinds (Westinghouse)
(Original Licensee Reference 2.17; Final Licensee Reference 10.3)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.20):

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

FRC has reviewed Reference 2.17, a description of Westinghouse tests performed on a prototype for the Brunswick plant electrical penetrations. A letter from W. Cahill of Consolidated Edison to B. Grier of NRC/OIE Region I states that Westinghouse advised Con Ed that the same tests are applicable to the Indian Point No. 3 penetrations. From the review of Reference 2.17, FRC notes the following:

a. The prototype was designed with three types of conductors: 36 #1 AWG, 50 #10 AWG, and 5 thermocouple pairs. During testing, 9 of the #1 AWG were loaded with and 3 of the #10 AWG were loaded with The penetration interior and outboard temperatures during testing stabilized at The test conditions were steam for 6 hours followed by a decay to These conditions completely envelop the LOCA temperature and pressure profiles submitted by the Licensee.

b. No mention was made of the materials of construction of the penetration; this information is necessary to ascertain if chemical spray and radiation exposure is required and if materials are subject to degradation by thermal aging. No thermal aging, chemical spray, or irradiation was included in the test. The requirements of the Guidelines with regard to these environments have not been satisfied.

c. Leak testing with helium was stated to have been accomplished successfully (but no limiting rate reported) before the thermal/steam exposure. No helium leak testing was reported for the penetration during or after exposure. The internal pressure was measured before and after exposure, and a small difference (after, compared to before) was attributed to a gage reading error. The report claims the equality of internal pressure before and after exposure as evidence that no leakage occurred during testing.

d. Insulation resistance values during thermal/steam exposure, but stayed well within acceptable values. One #10 AWG resistance declined to a lower value than the others and also volts. The report states that this is acceptable, since the actual operating voltage would be FRC concurs with this interpretation.

e. The Licensee has not established that the maximum electric currents that could occur under LOCA conditions (including short circuits) were represented in the current loadings used in the test program. Also, no descriptions of the various types of electrical penetrations in the plant were presented, so it is not possible to verify that the test specimen does adequately represent installed equipment items.

LICENSEE RESPONSE:

The design of the penetration is such that all mechanical joints are metal-to-metal, metal-to-ceramic, or metal-to-glass. No reliance is placed on organic compounds or potting componds of any type to effect a mechanical joint and/or leaktight seal.

In addition, the Brunswick Nuclear Power Plant penetration is constructed the same as Indian Point's with respect to its seals. Attached [Reference 10.3] is an environmental qualification report for the Brunswick Plant which includes irradiation of the various components. Treated steam was used in the test which has a higher conductivity than boron, also the components which are exposed to the containment environment are non-corrosive with respect to boron.

FRC EVALUATION:

FRC has reviewed the Licensee Response and the copy of the design approval test reports attached to the response [10.3]. FRC has also reanalyzed the initial submittal and the FRC comments provided in the DITER and has considered information acquired during this EEQ review program pertaining to electrical penetrations at other Westinghouse plants. As a result, FRC has the following comments:

a. There were two distinct designs of penetrations used in Westinghouse plants at the time the Indian Point plants were being constructed. One type, manufactured by Crouse-Hinds, consisted of glass or ceramic and metal brazed and welded to form the shell and pressure boundary. These penetrations are shown on Crouse-Hinds Drawings 0100044, 0100350, 0100253, 0100324, 0100252, 0100411, 0100251, 0100696, and 0100350.

   There was another type designed by Westinghouse which consisted of seals between a metal shell and the conductors (Westinghouse Drawings 2802 and 2803).

b. The report provided by the Licensee discusses radiation tests on ceramics, as well as various insulation potting compounds and epoxies.

c. While the testing described in Reference 10.3 envelops the pressure, temperature, and spray conditions shown in Appendix A for the LOCA conditions of that normal or short-circuit currents were applied nor discloses whether radiation was simultaneous or sequential.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because traceability between the installed units and those tested cannot be verified from the documents provided in the Licensee submittals. The Licensee should verify from installation records and drawings specifically which penetrations are installed in the plant. If any are the Westinghouse canister-type metal shell penetrations similar to those described in (a) above, performance and integrity under short-circuit current loadings must be established because there are nonsafety-related and/or unqualified electrical circuits connected to the penetrations. If the penetrations are metal/ceramic as described in (a) above, they would be assigned to NRC Category I.a or I.b.

4.6.8 Equipment Item No. 42B
Hydrogen Recombiner Located Within Containment
Westinghouse Electric Corporation
(Licensee Reference 2.1)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.24):

Reference 2.1 is a Westinghouse Topical Report that contains the results of qualification test programs for an igniter-exciter unit (BLA Part No. 43737, Rev. A, Serial No. 001), and a Westinghouse 2-hp, 3-phase, 60-Hz, 230/460 volt motor with insulation. It is stated in the report that this 2-hp motor is constructed in the same manner as the actual 15-hp motor used in the recombiner. FRC has reviewed Reference 2.1 and notes the following:

a. The qualification program for the igniter-exciter unit included a high pressure steam exposure, nuclear irradiation to 173 Mrd, and a 22-day steam exposure to simulate this component of a LOCA. Functional operation tests were each day.

The environmental parameters during this sequence of tests fully envelop those to which the actual equipment may be exposed except for two omissions: thermal aging and chemical spray.

The Guidelines require that 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 should be established and a program instituted to monitor performance and analyze failures to determine whether they are random or aging-induced. No analysis of the susceptibility of the materials to aging degradation has been provided, nor has a period of qualified life been established or documentation of an ongoing failure monitoring/analysis program been submitted.

The Guidelines require that equipment which is potentially exposed to chemical sprays must be qualified for this environment by either test or analysis. Documentation providing evidence that the performance for this equipment is satisfactory under chemical spray exposure conditions, or test that it is completely protected from contact with the spray, should be submitted.

b. The qualification program for the blower motor consisted of a 200 Mrd gamma irradiation, thermal pre-aging (stated to be equivalent to 40-years), and a 22-day steam/chemical spray exposure to simulate a LOCA.

While the environmental parameters appear to be satisfactory, three major concerns remain concerning this test program. No documentation has been provided to justify:

- the use of a 2-hp motor instead of the actual equipment item
- the test sequence (i.e., irradiation prior to thermal aging)
- the validity of the thermal aging as being equivalent to 40 years of installed service life.

c. No evidence to justify the exclusion of the igniter and temperature detector (the other electrical components in the recombiner) from this program has been provided. This omission should be corrected.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

The Licensee did not provide a response to the DITER. Another review of the Licensee's cited reference and submittal made in response to IE Bulletin 79-01 [6] raises the following concerns. Recombiner components such as the exhaust thermocouples, blower damper control solenoid, pressure switches, associated wiring, and any terminal blocks or splices will require qualification documentation, because their failure could jeopardize the recombiner's operation. Although the Licensee indicates in Reference 6 that some of these may have been tested, FRC finds no evidence of this testing in the references cited. These components are located inside the containment and will be exposed to the long-term accident environments. Because no chemical spray testing was performed on the recombiner unit, the Licensee should conduct an analysis or test to demonstrate that this service condition would not degrade the recombiner.

The Licensee has stated that the 15-hp Indian Point Unit No. 3 recombiner blower motor has the same construction as the 2-hp motor tested by Westinghouse. Evidence of this similarity should demonstrate that:

(1) the bearing system for the Indian Point Unit No. 3 plant motor is equivalent to or better than the 2-hp test motor's bearings

(2) the lubrication used in the motor can withstand the radiation and steam environment of the Indian Point Unit No. 3 containment

(3) the splices for the motor-lead and lead-to-cables for the Indian Point Unit No. 3 plant motor were identical or superior to those used in the tested unit.

Licensee Reference 2.1 (Westinghouse Report WCAP-7410-L, Vol. II) states that the expected life of the test motor's insulation is 7 years of continuous operation or 40 years of noncontinuous operation. This is expected to be the case for the Indian Point Unit No. 3 recombiner motor. The Licensee should establish the motor's overall qualified life in accordance with Section 4.1.3.

FRC CONCLUSION:

The overall hydrogen recombiner and blower motor unit is assigned to NRC Category V. Specific components of the recombiner will require qualification documentation to ensure that the overall recombiner will be operable for a long-term postulated accident environment. Additional evidence documenting traceability of the blower motors installed in the plant to the tested motor should be provided by the Licensee, together with an explicit determination of the unit's qualified life in accordance with Section 4.1.3.

4.6.9 Equipment Item No. 19
Solenoid Valves Located in the Auxiliary Pump Room
ASCO Model 8300
(Licensee Reference 2.4)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.11):

Licensee Reference 2.4 was discussed in the preceding subsection 3.3.2.10 [4.5.2.7 of this report], and this is the same as Equipment Item No. 25. The Licensee also notes that (1) these valves will fail to the "closed" (safe) position for all potential modes of failure; (2) the ambient environmental conditions in the locations where they are installed will not change significantly from the conditions existing during normal operation, when accidents occur; and (3) some of the valves are needed for only a period of a few minutes, while others are needed for a long time following an accident.

FRC has not had the opportunity to thoroughly review the plant arrangement and the "systems aspects" of the situation to verify the Licensee's analysis. Based upon a preliminary review, however, FRC has some concerns with regard to the possibility of a significant steam or water jet environment being present in these locations as a result of a steamline or feedline break.

In addition, as noted above, the following Guidelines requirements have not been addressed: aging degradation has not been considered; qualified life has not been established; and a program has not been established to ascertain whether any in-service failures during the installed life of the equipment are the result of aging degradation.

LICENSEE RESPONSE:

This concern is covered by the jet and water discussion in Section 3.3.2.5.

FRC EVALUATION:

As discussed in Section D.1 of Appendix D, the Licensee is relying upon an unqualified system to prevent a harsh environment in this area. Since there is no assurance that the area will be mild during a HELB, this equipment requires qualification for the environment to which it is subject.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because there is no evidence of qualification for the environment to which it can be exposed. The Licensee is relying upon an unqualified system to maintain a nonharsh environment in the auxiliary pump room. The Licensee should determine the qualified life of non-metallic parts based on manufacturers' recommendations, so that proper maintenance can be scheduled and performed.

4.6.10 Equipment Item Nos. 30 and 31
Limit Switches Located in the Steam/Feedline Penetrations Area and Pipe Penetrations Area
NAMCO Models SL3 and D2400
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.1.4 and 3.3.1.5):

The stated design basis event environment deviates only slightly from ambient conditions for this item. The Licensee states that the limit switch is for position indication only and that there is no known failure which would cause the valve to open. Provided that the NRC agrees that valve position indication is not a safety function, FRC finds that qualification is not required for this item.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

Although these limit switches provide position indication only, the function of the switches is basically to indicate proper shutting of the containment isolation valves (with certain exceptions). The closing of containment isolation valves upon receipt of a containment isolation signal requires reliable indication in order for the operator to know that the valves have performed their safety function. This is particularly true following a MSLB accident, when the position of the MSIVs may be critical to mitigating the accident and preventing complications with RCS pressure and volume control. Continued reliable position indication is also significant for the long term to prevent possible misinterpretation of valve status by the operators that could result in undesirable operator action. There is no information available to FRC which demonstrates any type of qualification for these items.

FRC CONCLUSION:

This item is assigned to NRC Category V because the valve position indication must be qualified for the environment to which the equipment may be exposed, and no evidence of such qualification has been provided by the Licensee (see Section D-7 of Appendix D).

4.6.11 Equipment Item Nos. 20 and 23A
Solenoid Valves Located in the Steam and Feedline Penetrations Area
20: Lawrence Model 110114W
Actuates Main Steam Isolation Valves
23A: ASCO Model 8316
Actuates AFW Pump Turbine Steam Supply Isolation Valves
(SOV-1310A, B)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.12):

The Licensee states that (i) Item No. 20 valves will fail to the "closed" (safe) position for all potential modes of failure; (ii) the ambient environmental conditions in the locations where they are installed will not change significantly from the conditions existing during normal operation, when accidents occur; and (iii) Item No. 20 valves are needed for only a few minutes, while the others are needed for only 30 minutes following an accident.

FRC has not had the opportunity to thoroughly review the plant arrangement and the "systems aspects" of the situation to verify the Licensee's analysis. Based upon a preliminary review, however, FRC has some concerns with regard to the possibility of a significant steam or water jet environment being present in these locations as a result of a steamline or feedline break and also with regard to the required times-for-functioning cited by the Licensee.

In addition, as noted above, the following Guidelines requirements have not been addressed: aging degradation has not been considered; qualified life has not been established; and a program has not been established to ascertain whether any in-service failures during the installed life of the equipment are the result of aging degradation.

LICENSEE RESPONSE:

This concern is covered by the jet and water discussion in Section 3.3.2.5 [4.5.2.4 of this report]. In addition, the Lawrence solenoid valves used on the main steam stop valves are discussed by the High Energy Line Analysis. The report states that the solenoids are protected from steam and water jetting by the shield wall.

FRC EVALUATION:

FRC believes that qualification is required for this equipment. As is noted in Section 4.1.2.2, FRC believes the HELB environmental conditions could be "harsh" rather than "mild" as the Licensee has stated. No evidence of qualification has been provided. The Guidelines requirements regarding aging degradation and qualified life should be addressed by the Licensee. The concerns expressed in Section 4.1.3 on these subjects are of particular importance for solenoid valves that are energized during normal plant operation.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because qualification documentation has not been provided. The Licensee should determine the qualified life of non-metallic parts based on the manufacturer's recommendations so that proper maintenance can be scheduled and performed.

4.6.12 Equipment Item No. 26
Solenoid Valves Located Within Containment
Skinner, Model Not Stated
Actuates Fan Cooler Unit Dampers (31, 32, 33, 34, 35)
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.1.2):

The Licensee states that these valves will close (their safe position) for all potential modes of failure and that they are automatically de-energized upon the occurrence of an SI signal. The same signal also isolates the instrument air supply line to the containment and trips the air compressor.

FRC has not had the opportunity to verify the Licensee's analysis of the "systems aspects" of the situation, since no supporting documentation has been provided. If a subsequent review of the systems aspects does provide this verification, FRC would agree that qualification of these solenoid valves is not required.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

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 plus expected operating time under LOCA conditions to verify proper operation (see Section 2.2.4 of this report). (This requirement was established subsequent to the preparation of the DITER.) The effects of the normal service environment on the equipment should be evaluated and the qualified life explicitly determined.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because no evidence of qualification has been provided.

4.7 NRC Category VI
EQUIPMENT FOR WHICH QUALIFICATION IS DEFERRED

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

4.7.1 Equipment Item No. 32A
Limit Switches Located in the Steam/Feedline Penetration Areas
Micro Switch Model EXAR 7313
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.1.6):

The stated design basis event environment deviates only slightly from ambient conditions for this item. The Licensee states that the limit switch is for position indication only. Provided that the NRC agrees that valve position indication is not a safety function, FRC finds that qualification is not required for this item.

LICENSEE RESPONSE:

[No response provided in Reference 9.]

FRC EVALUATION:

Although the isolation valves in the steam supply to the AFW pump turbine are not containment isolation valves, they do perform a safety function by limiting the severity of the environment in the auxiliary pump room following a HELB to the steam supply, thereby protecting a large amount of safety-related equipment. Consequently, the indication that these valves have performed their function is of safety significance. Since the valves and switches complete their function before the environment becomes harsh, qualification review of these switches is deferred until after February 1, 1981 in accordance with Section 2.2.3 of this report.

FRC CONCLUSION:

This item is assigned to NRC Category VI because, since the environment is mild for the accident the valves mitigate, qualification may be deferred in accordance with Section 2.2.3.

4.7.2 Equipment Item Nos. 34B and 34C
Large Electric Motors Located in the Primary Auxiliary Building
34B: Westinghouse, 509 US Frame
SI Pump Drive
34C: Westinghouse, 509 UPZ Frame
RHR Pump Drive
(Licensee Reference 2.10)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT (3.3.2.16):

The cited reference is a generic report for environmental qualification of Class 1E motors supplied by Westinghouse for outside containment use in nuclear power plants. FRC's comments concerning this reference are:

a. The report states that only stator winding insulation and motor bearings will show effects of environmental aging. It further states that since the same insulation system and only two different bearings were used in all motors, it is possible to perform generic qualification. FRC finds this position to be acceptable. However, from the information presented in the report, it is not possible to determine what models of electric motors have been generically qualified or what the actual test specimen was. 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.

b. Aging of the specimen is considered in the test program, and the nuclear radiation exposure was much greater than the expected exposure for this equipment. However, the program did not consider the consequences of increased ambient temperatures. An explicit determination of qualified life should be made.

LICENSEE RESPONSE:

We are obtaining further documentation from Westinghouse to substantiate the conclusions that the motors are qualified under the documentation in our submittal dated April 28, 1980.

FRC EVALUATION:

The comments made in the DITER continue to apply since no further documentation was provided by the Licensee.

The Licensee should demonstrate qualification for such items as the installed motor's bearings, lubrication, splices, and insulation by comparing them to previously tested motors and components. As discussed in the Methodology, Section 4.1 of this text, the exact accident environmental service conditions require additional review in order to confirm this area as nonharsh.

FRC CONCLUSION:

These equipment items are assigned to NRC Category VI because the area has been defined as nonharsh by the Licensee except for the radiation service condition that occurs during the long-term cooldown phase. Because these equipment items are located in a mild area for the accident they mitigate or are needed for cold shutdown, they are deferred in accordance with Sections 2.2.3 and 2.2.5. The Licensee should establish traceability of the motors to previous testing and should address the equipment's qualified life and aging mechanisms.

4.8 SUMMARY OF THE EVALUATION

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

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

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

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

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

O = Assignment to an NRC qualification category.

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

Table 4-1

NUMBER OF EQUIPMENT ITEMS IN EACH QUALIFICATION CATEGORY

NRC Category No.	Category Definition	Number of Equipment Items
I.a	Equipment Fully Satisfies All Applicable Requirements for the Life of the Plant	0
I.b	Equipment Does Not Meet All Applicable Requirements; However, Deviations are Judged Acceptable for the Life of the Plant	0
II.a	Equipment Satisfies All Applicable Requirements With the Exception of Qualified Life	7
II.b	Equipment Satisfies All Applicable Requirements With the Exception of Qualified Life Provided That Specific Modifications are Made	3
II.c	Equipment Does Not Meet All Applicable Requirements; However Deviations Are Judged Acceptable With the Exception of Qualified Life	10
III	Equipment is Exempt from Qualification Requirements	2
IV.a	Equipment has Qualification Testing Scheduled	0
IV.b	Equipment has High Likelihood of Operability; However, Proper Qualification Documentation Has Not Been Made Available for Review	22
V	Equipment is Unqualified	14
VI	Equipment Qualification is Deferred	3
		61

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Table 4-2

Franklin Research Center A Division of The Franklin Institute The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK 02. 206	REACTOR TYPE PWR	PLANT NAME Indian Point 3	PAGE 1
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Power Authority of the State of New York	
SEP PLANTS	DOCKET 50-286	NRC TAC 12974	DATE/ENGINEER 4-6-81 Ja	

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER																		
	1	2	3	4A	4B	5A	5B	6	7	8	9	10	11A	11B	12	13	14A	14B	14C
GUIDELINE REQUIREMENTS.	(DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)																		
EVIDENCE OF QUALIFICATION												X							
RELATIONSHIP TO TEST SPECIMEN	L	L	L	L	L	L	L	X	L	L	L		X	X	X	X	X	X	X
AGING DEGRADATION EVALUATED	L	L	L	L	L	L	L	X	L	L	L		L	L	L	L	L	L	L
QUALIFIED LIFE ESTABLISHED	L	L	L	L	L	L	L	X	L	L	L		L	L	L	L	L	L	L
PROGRAM TO IDENTIFY AGING													L	L	L	L	L	L	L
QUAL. FOR STEAM EXPOSURE															L				
PEAK TEMPERATURE ADEQUATE																			
PEAK PRESSURE ADEQUATE																			
TEST DURATION ADEQUATE								X											
REQUIRED PROFILE ENVELOPED																			
QUAL. FOR SUBMERGENCE																			
QUAL. FOR CHEMICAL SPRAY																			
QUAL. FOR RADIATION																			
BETA RADIATION CONSIDERED																			
TEST SEQUENCE													X	X	X				
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	O	O	O	O	O	O	O		O	O	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							
VI. QUAL. IS DEFERRED																			
REPLACEMENT SCHEDULE								R				R	R	R	R				
	MVA	MVA	MVA	MVA	MVA	MVA	MVA	MVA	MVA	MVA	MVA	LS	T	T	PT	PT	PT	T	PT

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Table 4-2 (Cont.)

Franklin Research Center, A Division of The Franklin Institute, The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK 02. 206	REACTOR TYPE PWR	PLANT NAME Indian Point 3	PAGE 2
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Power Authority of the State of New York	
SEP PLANTS	DOCKET 50-286	NRC TAC 12974	DATE/ENGINEER 4-6-81 ga	

SUMMARY REVIEW	15	16A	16B	17	18A	18B	19	20	21	22	23A	23B	24	25	26	27	28A	28B	29
GUIDELINE REQUIREMENTS, (DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)																			
EVIDENCE OF QUALIFICATION				X			X	X	X	X	X		X	X	X				X
RELATIONSHIP TO TEST SPECIMEN	X	X	X																
AGING DEGRADATION EVALUATED	L	L	L													L			
QUALIFIED LIFE ESTABLISHED	L	L	L		L											L	L	L	
PROGRAM TO IDENTIFY AGING	L	L	L													L	L	L	
QUAL. FOR STEAM EXPOSURE			L																
PEAK TEMPERATURE ADEQUATE						X													
PEAK PRESSURE ADEQUATE						X													
TEST DURATION ADEQUATE																			
REQUIRED PROFILE ENVELOPED																			
QUAL. FOR SUBMERGENCE																			
QUAL. FOR CHEMICAL SPRAY						X													
QUAL. FOR RADIATION																			
BETA RADIATION CONSIDERED																			
TEST SEQUENCE			X																
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																O			
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					
V. EQUIP. NOT QUALIFIED				O		O	O	O			O				O				O
VI. QUAL. IS DEFERRED																			
REPLACEMENT SCHEDULE			R	R															
	FT	PT	PT	FT	SV	SV	SV	SV	SV	SV	SV	SV	SV	SV	SV	SV	XS	XS	XS
								△1			△1	△1			△1	△1			

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Table 4-2 (Cont.)

Franklin Research Center A Division of The Franklin Institute The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK 02. 206	REACTOR TYPE PWR	PLANT NAME Indian Point 3	PAGE 3
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Power Authority of the State of New York	
SEP PLANTS	DOCKET 50-286	NRC TAC 12974	DATE/ENGINEER 4-6-81 Ja	

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER																		
	30	31	32A	32B	33	34A	34B	34C	35	36	37A	37B	37C	38	39	40A	40B	41	42A
GUIDELINE REQUIREMENTS,	(DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)																		
EVIDENCE OF QUALIFICATION	X	X	X			X			X	X									
RELATIONSHIP TO TEST SPECIMEN									X	X					X				
AGING DEGRADATION EVALUATED				L		X			X	X	L	L							L
QUALIFIED LIFE ESTABLISHED	X	X		L	L	X			X	X	L	L		X	X				L
PROGRAM TO IDENTIFY AGING	X	X		L	L	X			X	X	L	L		X					L
QUAL. FOR STEAM EXPOSURE						X								X					
PEAK TEMPERATURE ADEQUATE	X	X				X				X									
PEAK PRESSURE ADEQUATE	X	X																	
TEST DURATION ADEQUATE																			
REQUIRED PROFILE ENVELOPED	X	X																	
QUAL. FOR SUBMERGENCE					X											X	X	X	
QUAL. FOR CHEMICAL SPRAY										X				X					
QUAL. FOR RADIATION										X				X		X	X	X	
BETA RADIATION CONSIDERED																X	X	X	
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				O															O
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	
V. EQUIP. NOT QUALIFIED	O	O				O								O	O				
VI. QUAL. IS DEFERRED			O				O	O											
REPLACEMENT SCHEDULE																			
	XS	XS	XS	XS	XS	M	M	M	M	M	E/P	E/P	E/P	TS	P	C	C	C	P
	Δ	Δ	Δ	Δ						Δ									Δ

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Table 4-2 (Cont.)

Franklin Research Center, A Division of The Franklin Institute, The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK: 02 206	REACTOR TYPE: PWR	PLANT NAME: Indian Point 3	PAGE: 4
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT: 02G-C5257-01		UTILITY: Power Authority of the State of New York	
SEP PLANTS	DOCKET: 50-286	NRC TAC: 12974	DATE/ENGINEER: 4-6-81 / 9a	

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER: 42B	43	44	45
GUIDELINE REQUIREMENTS. (DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)				
EVIDENCE OF QUALIFICATION	X			
RELATIONSHIP TO TEST SPECIMEN	X			
AGING DEGRADATION EVALUATED	X	X	L	
QUALIFIED LIFE ESTABLISHED	X	X	L	
PROGRAM TO IDENTIFY AGING	X			
QUAL. FOR STEAM EXPOSURE				
PEAK TEMPERATURE ADEQUATE		X		
PEAK PRESSURE ADEQUATE				
TEST DURATION ADEQUATE				
REQUIRED PROFILE ENVELOPED				
QUAL. FOR SUBMERGENCE				
QUAL. FOR CHEMICAL SPRAY	X			
QUAL. FOR RADIATION	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	
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		
V. EQUIP. NOT QUALIFIED	O			
VI. QUAL. IS DEFERRED				
REPLACEMENT SCHEDULE				

HR RTD

5. CONCLUSIONS

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

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

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

The present assessment of the status of environmental qualification of the safety-related electrical equipment installed in Indian Point Unit No. 3 involves only equipment located in the "harsh environment" areas and needed to ensure hot shutdown of the plant. The EEQ review of equipment items located in "mild" areas and equipment needed for TMI Action Plan compliance has been deferred by the Licensee until after February 1, 1981.

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APPENDIX A - ENVIRONMENTAL SERVICE CONDITIONS

This appendix contains a summary of the information concerning expected environmental service conditions in various locations within the plant. Figure A-1 shows the plant arrangement and serves to define specific buildings and other locations. The specific environmental service conditions corresponding to different plant locations that were used in this technical evaluation are stated in this appendix, based upon the information presented in the Licensee's submittal [1].

As noted in Section 4.1.2 of this report, only environments resulting from HELB accidents inside or outside containment are considered in this review. It is also noted in Section 4.1.2 that FRC questions whether some of the environmental parameters are sufficiently conservative. The temperatures in two locations (auxiliary pump room and steam feedline penetrations) have been assumed to be clearly "harsh" (the Licensee implies in Reference 1 that the conditions are not harsh even though HELBs occur in these areas).

Environment "C" - Inside Reactor Containment

Normal Operation

Temperature	120°F (maximum)
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Not stated

Accident Conditions

For PWR plants, the DOR Guidelines state that the environmental service conditions inside containment for the most severe LOCA be established by the Licensee based on the FSAR analysis. In addition, for plants equipped with

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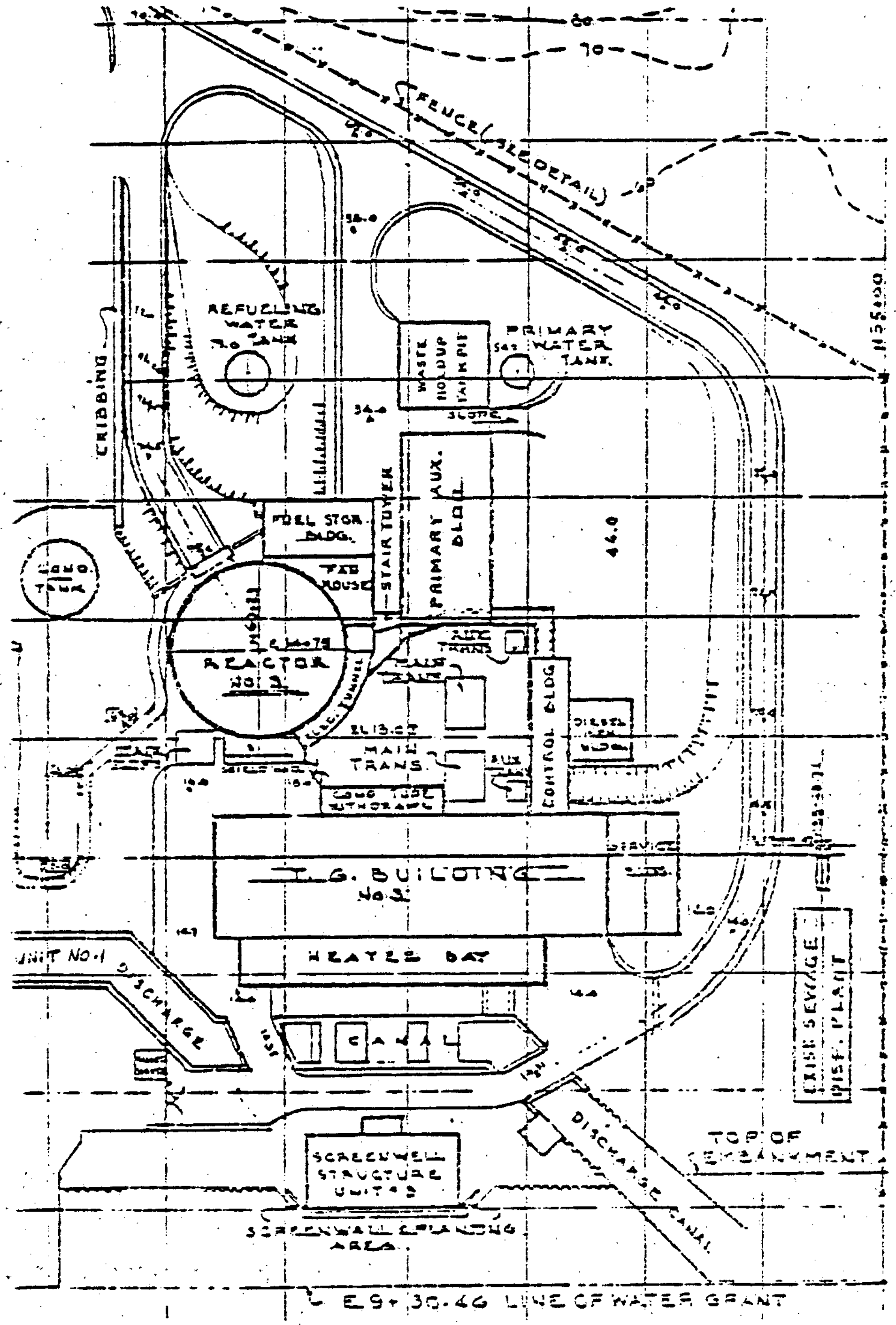

Figure A-1. Identification of Individual Buildings and Specific Areas at Indian Point No. 3 Plant. (Note: Unit Nos. 1, 2, and 3 share the same site.)

FIGURE SUPPLIED BY THE LICENSEE

automatic containment spray systems not subject to single component failure or delayed initiation, the Guidelines state that equipment qualified for the most severe LOCA environment is also considered qualified for the postulated MSLB. The design of the Indian Point Unit No. 3 nuclear power plant satisfies these criteria. The environmental conditions resulting from a feedline break are less severe than those from a LOCA.

The NRC has made an independent assessment of the short- and long-term temperature profiles within the containment and has concluded that the conditions stated by the Licensee are acceptable for the purposes of this accelerated environmental qualification review [11]. This reference also notes that the NRC has calculated somewhat higher peak conditions (268°F/44 psig compared to 258°F/40 psig), and "the Licensee should be cautioned that some margin in its qualification effort would be prudent."

The environmental parameters used for the assessment of qualification of equipment inside the containment are:

Temperature	Figure A-2
Pressure	Figure A-3
Humidity	100% (nominal)
Spray	Solution of boric acid (2000 ppm of boron) plus 40% sodium hydroxide in water (pH = 10)
Radiation	20 Mrd*
Flooded Depth	**

*The Licensee has stated that the value suggested in the DOR Guidelines document has been used. This does not include the contribution from beta radiations.

**The Licensee has stated that the only safety-related equipment that will become submerged is electrical power and control cables.

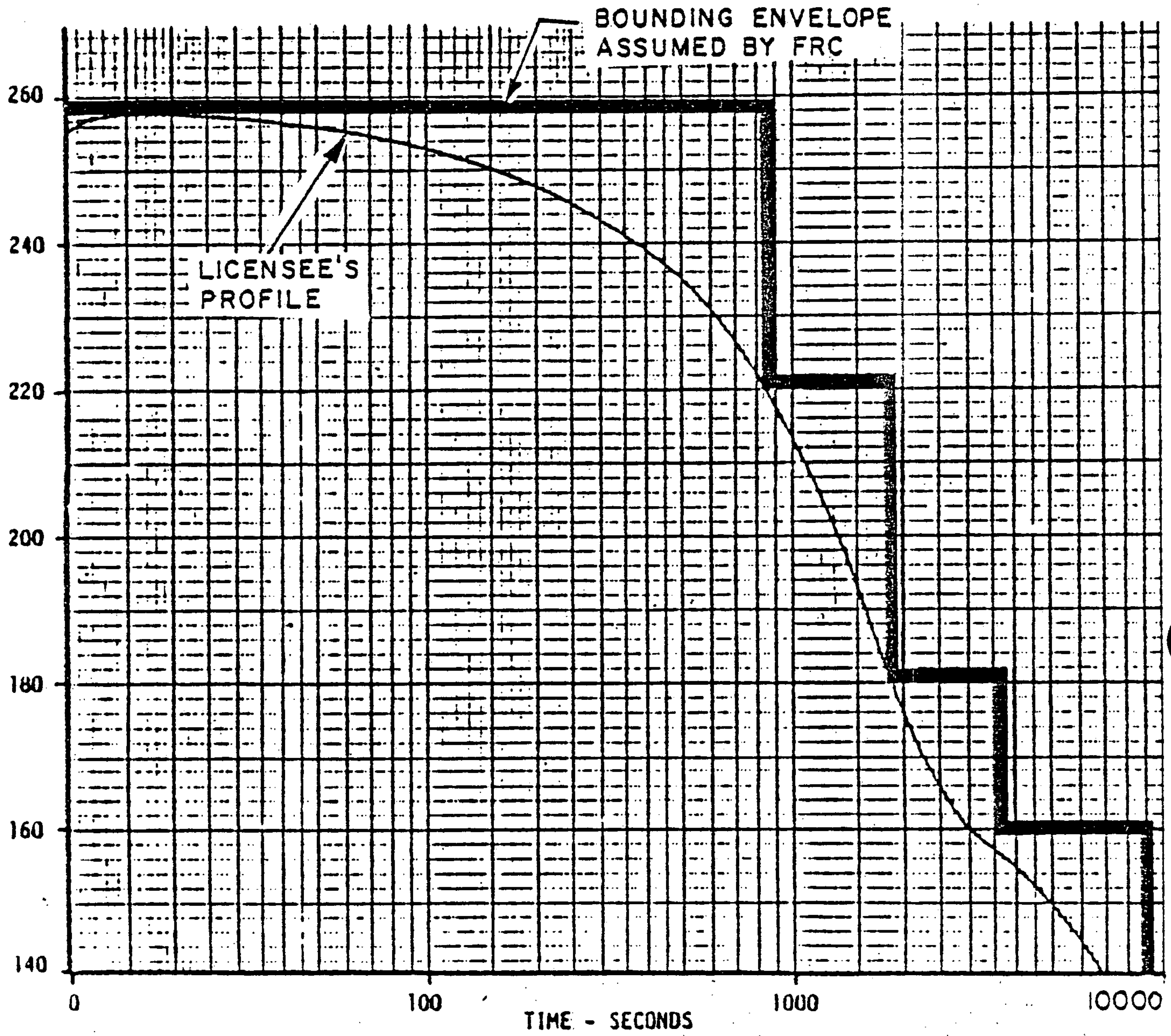

Figure A-2. LOCA Condition Temperature Vs. Time Profile Within Containment [1]

FIGURE SUPPLIED BY THE LICENSEE

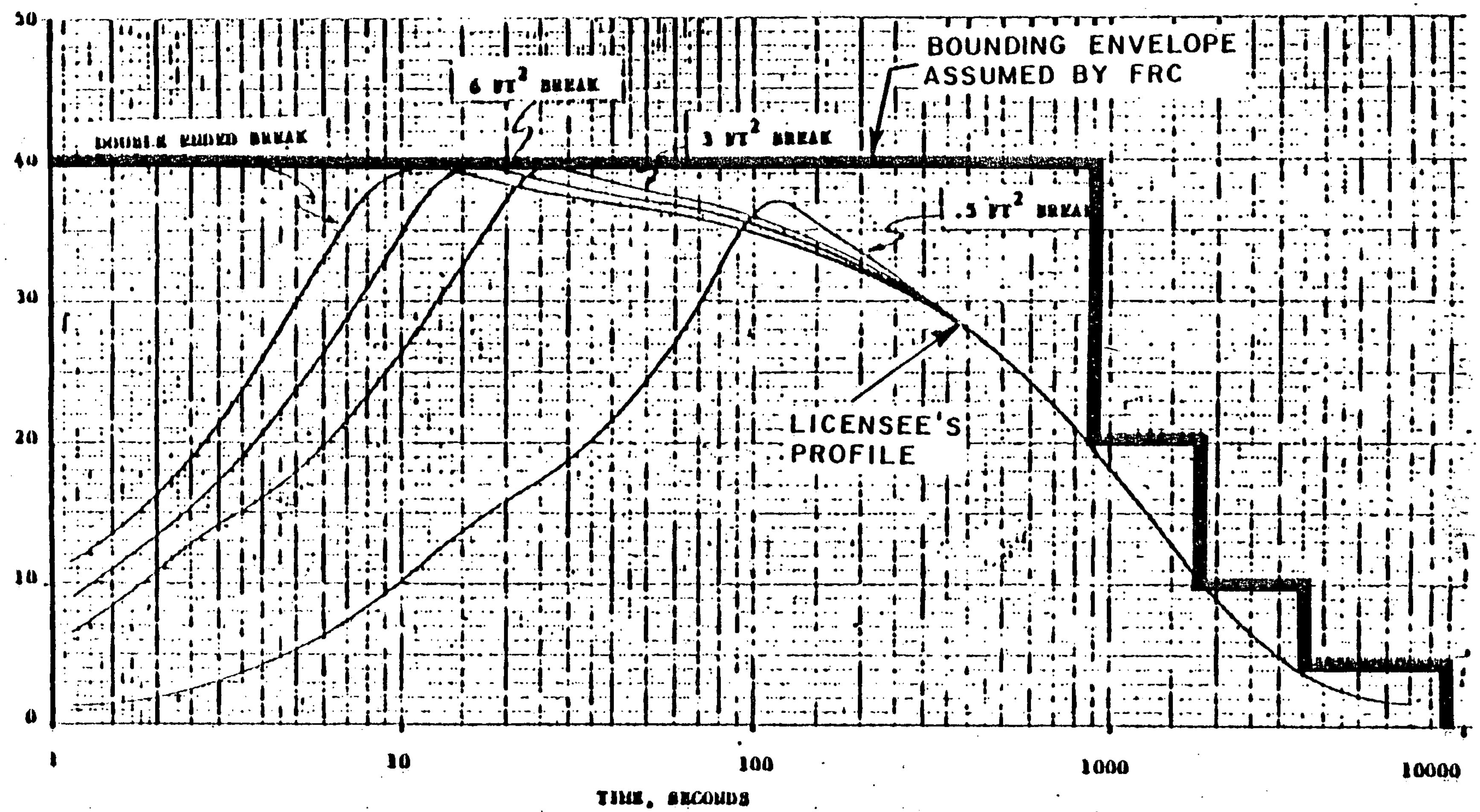

Figure A-3. LOCA Condition Pressure Vs. Time Profile Within Containment [1]

FIGURE SUPPLIED BY THE LICENSEE

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Environment "RH" - Residual Heat Removal Pump Area in PAB

Normal Operation*

(When the reactor is shut down--assumed to be 15% of time; otherwise, conditions are 50-104°F/0 psig/50% RH/negligible radiation.)

Temperature	100°F
Pressure	0 psig
Humidity	65% RH
Radiation	∿ 1 rd/h

Accident Conditions

Temperature	No increase from normal
Pressure	No increase from normal
Humidity	No increase from normal
Radiation	3.6 Mrd integrated dose (max.); values dependent upon specific locations
Spray	Not stated
Flooded Depth	No submergence

Environment "AP" - Auxiliary Pump Room

Normal Operation*

Temperature	50-104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible

Accident Conditions

Temperature	213°F for a few minutes; reduced to pre-accident conditions within 20 min**
Pressure	0.9 psig for 20 min**
Humidity	100% for 20 min**
Radiation	Negligible
Spray	Not stated
Flooded Depth	No submergence

*The Licensee has not stated the environmental parameters corresponding to normal plant operation. FRC has assumed these values.

**The Licensee stated 5-min duration and "ambient" humidity but did not demonstrate that these valves are conservative. FRC has assumed the 20-min and 100% RH values, but does not know whether they are sufficiently conservative (see Section 4.1.2.2).

Environment "SP" Steam and Feedline Penetrations Area

Normal Operation*

Temperature	50-104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible

Accident Conditions

Temperature	213°F for 20 min**
Pressure	0.42 psig for 20 min** (time not stated)
Humidity	100%**
Radiation	Negligible
Spray	Not stated
Flooded Depth	No submergence

Environment "PP" and "SI" - Pipe Penetrations Area Adjacent to PAB ("PP") and Safety Injection Area in PAB ("SI")

Normal Operation*

Temperature	50-104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible except near RHR piping during plant shutdown

Accident Conditions

Temperature	No increase from normal***
Pressure	No increase from normal***
Humidity	No increase from normal***
Radiation	3.6 Mrd integrated dose (max.); values dependent upon specific locations
Spray	Not stated
Flooded Depth	No submergence

*The Licensee has not stated the environmental parameters corresponding to normal plant operation. FRC has assumed these values.

**The Licensee has stated "negligible temperature increase and a pressure of 0.42 psig," and "ambient" humidity, but does not present any information regarding time. FRC has assumed the 20-min value, but does not know whether this is sufficiently conservative (see Section 4.1.2.2).

***FRC has used information from Reference 1 in this EEQ review, but has not verified that the temperature does not increase beyond the range experienced during normal plant operation.

APPENDIX B - LISTING OF SAFETY-RELATED ELECTRICAL EQUIPMENT

The following table lists the groupings of safety-related electrical equipment for the Indian Point Unit No. 3 nuclear power plant. Equipment item numbers provided in the table are used in the Equipment Environmental Qualification Summary Forms and in the equipment qualification discussions presented in Section 4. The listing includes identification of manufacturer, model designation, plant location, time needed during the accident and post-accident period, and cited qualification references, all as shown by the Licensee in References 1 and 9. The designation "long" indicates that the Licensee stated "30 days."

The following abbreviations have been used to designate location:

C = Inside containment
PAB = Primary auxiliary building
PP = Pipe penetration areas outside of containment adjacent to PAB
AP = Auxiliary pump room
SP = Steam and feedline penetration area
SI = Safety injection pump room in PAB
RH = Residual heat removal pump room in PAB

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	QUALIFICATION REFERENCES
1	Motorized Valve Actuator Limitorque SMB-00 (H)	C	Long	2.1, 2.2, 2.20, 2.21, 10.1
2	Motorized Valve Actuator Limitorque SMB-0 (H)	C	Long	2.1, 2.2, 2.20, 2.21, 10.1
3	Motorized Valve Actuator Limitorque SMB-3 (H)	C	Long	2.1, 2.2, 2.20, 2.21, 10.1
4A	Motorized Valve Actuator Limitorque SMB-0 (B)	PP	Long	2.1, 2.19
4B	Motorized Valve Actuator Limitorque SMB-0 (B)	SI	Long	2.1, 2.19
5A	Motorized Valve Actuator Limitorque SMB-00 (B)	PP	Long	2.1, 2.19
5B	Motorized Valve Actuator Limitorque SMB-00 (B)	SI	Long	2.1, 2.19
6	Motorized Valve Actuator Limitorque SMB-00 (B)	C	Intermediate (< 8 h)	2.1, 2.19, 10.1
7	Motorized Valve Actuator Limitorque SMB-2 (B)	C	Intermediate (< 8 h)	2.1, 2.19, 10.1

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ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	QUALIFICATION REFERENCES
8	Motorized Valve Actuator Limitorque SMB-000 (B)	SI	Long	2.1, 2.19
9	Motorized Valve Actuator Limitorque SMB-1 (B)	PP	Long	2.1, 2.19
10	Sump Level Switch GEMS LS 1900	C	Short (30 min)	2.18
11A	Pressurizer Level and Steam Generator Level Transmitters Foxboro E13DM (MCA)	C	Short (5 min)	2.3, 2.6
11B	High Head and Spray Flow Transmitters Foxboro E13DM (MCA)	C	Short (30 min)	2.3, 2.6
12	RCS Pressure Transmitter Foxboro E11GH	C	Short (5 min)	2.3, 2.6
13	Containment Pressure Transmitter Foxboro E11GM	PP	Short (5 min)	2.6
14A	Pressure Transmitters Foxboro E11GM	AP	Long	None

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	QUALIFICATION REFERENCES
14B	Main Steam Pressure and FW Pressure Transmitters Foxboro E11GM	AP	Short (5 min)	2.7
14C	Feedwater Flow Transmitter Foxboro E11DM	AP	Short (5 min)	2.7
15	Aux. FW Flow Transmitters Foxboro E13DM	AP	Long	None
16A	SI Pump Pressure Transmitter Foxboro E11GM	SI	Long	2.6
16B	Pressurizer Pressure Transmitter Foxboro E11GM (MCA)	C	Short (5 min)	2.3, 2.6
17	RHR Flow Transmitter Barton 386	C	Short (30 min)	2.1, 2.3
18A	Solenoid Valve ASCO NP-8316	PP	Long	2.8
18B	Solenoid Valve ASCO NP-8316	C	Long	2.8
19	Solenoid Valve ASCO 8300	AP	Long	None
20	Solenoid Valve Lawrence 110114W	SP	Short (5 min)	None

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	QUALIFICATION REFERENCES
21	Solenoid Valve ASCO 8314	PP	Long	2.4, 2.5
22	Solenoid Valve ASCO 8316	PP	Long	2.4, 2.5
23A	Solenoid Valve ASCO 8316	SP	Short (30 min)	None
23B	Solenoid Valve ASCO 8300	SP	Short (5 min)	2.4
24	Solenoid Valve ASCO 8317	PP	Long	2.4, 2.5
25	Solenoid Valve ASCO 8300	PP	Long	2.4, 2.5
26	Solenoid Valve Skinner	C	Long	None
27	Solenoid Valves Lawrence 629BC85PS	PP	Long	None
28A	Position Switch NAMCO EA-180	C	Long	2.9, 10.2
28B	Position Switch NAMCO EA-180	PP	Long	2.9, 10.2
29	Position Switch NAMCO EA-170	AP	Long	None
30	Position Switch NAMCO SL 3	SP	Short (30 min)	None

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	QUALIFICATION REFERENCES
31	Position Switch NAMCO D2400X	PP	Long	None
32A	Position Switches Micro-Switch EXAR-7313	SP	Short (30 min)	None
32B	Position Switch Micro-Switch EXHAR-3	PP	Long	None
33	Position Switch NAMCO EA-740	C	Long	2.9
34A	AFW Pump Motors Westinghouse 509 US	AP	Long	None
34B	SI Pump Motors Westinghouse 509 US	SI	Long	2.10
34C	RHR Pump Motors Westinghouse 509 UPZ	RH	Long	2.10
35	SI Recirculation Pump Motors Westinghouse 588-5	C	Long	2.11, 2.12
36	Fan Cooler Motors Westinghouse Lifeline 69F97009	C	Long	2.11, 2.12
37A	E/P Converters Fisher Type 546	AP	Long	None
37B	E/P Converters Fisher Type 546	SP	Short (30 min)	2.13

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	LOCATION	TIME REQUIRED	QUALIFICATION REFERENCES
37C	E/P Converters Fisher Type 546	PP	Long	None
38	Terminal Blocks Westinghouse 542247 (805432)	C	Long	2.14, 2.15, 2.16
39	Electrical Penetrations Westinghouse	C	Long	2.17, 10.3
40A	Power Cables/Splices, Silicone Rubber Ins. Asbestos Braid Jacket/ Raychem	C	Long	2.1, 2.22, 2.23
40B	Power Control Cables/Splices Kerite/Raychem	C	Long	2.1, 2.22, 2.23
41	Instrument Cables/Splices Manufacturer not identified/ Raychem	C	Long	2.24
42A	Hydrogen Recombiner Panel Westinghouse	PP	Long	None
42B	Hydrogen Recombiner Westinghouse	C	Long	2.1
43	Resistance Temperature Detectors Sostman 11901B	C	Short (5 min)	2.18
44	Connectors Conax Model N11001-33	C	Long	10.4
45	Position Switch NAMCO D2400X	PP	Long	None

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

The NRC transmitted to the Licensees for the SEP plants, Indian Point Units No. 2 and 3, and Zion Units 1 and 2 the DOR Guidelines for evaluating Class 1E equipment qualification and the "Guidelines for Identification of That Safety Equipment of SEP Operating Reactors for Which Environmental Qualification Is To Be Addressed." Based on these documents, the Licensee has submitted a list of safety-related systems that must function in order to mitigate the consequences of a design basis accident. Discussions between the Licensee and the NRC resulted in the following list of systems and display instrumentation for which the Licensee and the NRC have determined that qualification is to be addressed.

A. Safe Shutdown Systems

Reactor Protection System*
Residual Heat Removal System (including hot leg suction)++
Auxiliary Feedwater System*
Component Cooling Water System
Service Water System
Radiation Monitoring System and Sampling*
Emergency Diesel System*
480 V Switchgear System*
Motor Control System*
125 V dc System*

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

Pressurizer Pressure Relief+
Actuation System Safeguards
Containment Isolation System
Steam Line Isolation System
Feedwater Isolation System
Accumulator System
High Head Safety Injection System
Low Head Safety Injection System
Containment Spray System
Fan Cooler System
Hydrogen Recombiner System
Primary Auxiliary Building Ventilation System+++
Control Building HVAC System+++
Diesel Room Ventilation System+++

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*Systems which function for both safe shutdown and accident mitigation.
+To be added as "TMI Lessons Learned" requirement.
++System required for cold shutdown only.
+++The review of these systems has been deferred until after February 1, 1981, as stated in Section 2.2.3.

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

Pressurizer Pressure
RCS Pressure
Pressurizer Level
Steam Generator Level
Main Steam Pressure
Auxiliary Feedwater Flow
Containment Pressure**
Containment Sump Level**
RWST Level
CST Level
High-Head SI Flow**
Recirculation Spray Flow**
RHR Recirculation Flow
SI Pump Suction and Discharge Pressure**
Component Cooling Water Flow
Service Water Flow
Diesel Generator Monitoring

**Instruments required only for accident mitigation.

APPENDIX D - EVALUATIONS OF LICENSEE EXPLANATIONS OF ADEQUACY OF EQUIPMENT BASED ON SYSTEM OPERATIONAL CONSIDERATIONS

In the submittals from PASNY for the Indian Point Unit No. 3 plant [1,9], the Licensee presented various system operational reasons for classifying the environmental qualification of certain equipment items as satisfactory or not required. These reasons include the availability of redundant items (qualified and/or unqualified), the time of operation, and the need for the involvement of the equipment in specific design basis accidents.

At the request of the NRC, FRC has evaluated these Licensee explanations. The results of these evaluations are presented in this appendix. In many cases, the conclusions have also been included in the applicable sections of the text.

D.1 AUXILIARY FEEDWATER SYSTEM (AFW) EQUIPMENT IN THE AUXILIARY PUMP ROOM

Equipment Item No. 14A: AFP Suction and Discharge Pressure Transmitters
TDAFP Steam Supply Pressure Transmitter
City Water Supply Pressure Transmitter
Equipment Item No. 15: Auxiliary Feedwater Flow Transmitters
Equipment Item No. 19: City Water Suction Control Valves (PCV-1187, 1188, and 1189)
AFW Recirculation Flow Trip Solenoid Valve (FCV-1121, 1123)
AFW Pump Steam Pressure Control Valve (PCV-1139)
Equipment Item No. 29: AFW Recirculation Flow Trip Limit Switches
Equipment Item No. 34A: Auxiliary Feedwater Pump Motors

LICENSEE POSITION:

The Licensee states that this equipment remains in a normal environment following a loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, and main feed line break (MFLB) accident. The Licensee also states that the equipment remains in a near-normal environment following a high energy line break (HELB) in the steam supply to the auxiliary feedwater (AFW) pump turbine, except for a brief (i.e., minutes) excursion to 213°F. Following this HELB, area temperature and pressure return to normal within 5 minutes because sensors set at 135°F cause isolation of the steam line; this also precludes the need for AFW system operation as a result of the HELB.

FRC EVALUATION:

The Licensee has installed two independent isolation valves, in series, in the auxiliary steam supply line to the AFW pump turbine. These valves are located outside the auxiliary pump room and are signaled to close by independent temperature switches within the room, set to actuate at 135°F. This installation is designed to minimize the severity of the environment in the auxiliary pump room, the effects of steam jet impingement on equipment, and other consequences of a HELB in the pump room.

FRC stated in Section 4 of the DITER [16]:

The Licensee's justification for there being only a small temperature excursion in the event of a steam line break in the auxiliary pump room is that a temperature sensor will initiate a signal to isolate the line. Therefore, the sensor, wiring, and controls in this circuit should be added to the list of equipment items that are relied upon.

The Licensee has not yet responded to this concern. Therefore, it is not possible for FRC to agree that the environment in the auxiliary pump room will not change following a steam line break in the room.

Of the various AFW system equipment items located in the auxiliary pump room, the instrumentation is probably the most susceptible to functional impairment. This instrumentation is important to the proper operation of the AFW system. NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, has recently placed additional emphasis on the ability to monitor AFW system performance, including a recommendation that safety-grade auxiliary feedwater flow indication be provided. It is significant to note, however, that the accident which creates an abnormal environmental condition for this equipment (the HELB of the steam supply line) is terminated without the need to initiate AFW. Should AFW eventually be needed for a plant cooldown, the cooldown can be conducted using local AFW instrumentation and steam generator level instruments, provided that the motor-driven pumps remain functional.

FRC CONCLUSION:

Environmental qualification of AFW system instrumentation and other equipment in the auxiliary pump room is required whether or not the automatic isolation feature is demonstrated to be reliable, in order to ensure the availability of basic cooldown capability.

D.2 NON-AFW SYSTEM EQUIPMENT IN THE AUXILIARY PUMP ROOM

Equipment Item No. 14B: Main Steam and Steam Generator Feedwater Pressure Transmitters
Equipment Item No. 14C: Main Feedwater Flow Transmitters

LICENSEE POSITION:

The Licensee states that this equipment remains in a normal environment following a loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, and main feed line break (MFLB) accident. The Licensee also states that the equipment remains in a near-normal environment following a high energy line break (HELB) in the steam supply to the auxiliary feedwater (AFW) pump turbine except for a brief (i.e., minutes) excursion to 213°F. Following this HELB, area temperature and pressure return to normal within 5 minutes because sensors set at 135°F cause isolation of the steam line. This also precludes the need for AFW system operation as a result of the HELB.

FRC EVALUATION:

The Licensee has installed two independent isolation valves, in series, in the auxiliary steam supply line to the AFW pump turbine. These valves are located outside the auxiliary pump room and are signaled to close by independent temperature switches within the room, set to actuate at 135°F. This installation is designed to minimize the severity of the environment in the auxiliary pump room, the effects of steam jet impingement on equipment, and other consequences of a HELB in the pump room.

The pressure transmitters provide information to actuate safety injection and reactor trip and to shut the main steam or feedwater isolation valves (MSIVs) in case of an MSLB/MFWLB accident. The Licensee has implied that these instruments are not required to function following a HELB in the auxiliary pump room because of the automatic isolation feature discussed previously. It is noted that if main steam line isolation were signaled, it would not mitigate the consequences of a HELB to the AFW pump turbine since the AFW pump steam supply branches from the main steam lines upstream of the MSIVs. Therefore, it is important that the temperature trip for the AFW line function as designed to minimize the environmental effect to the steam and feedwater pressure transmitters and the main feedwater flow transmitter.

FRC CONCLUSION:

Environmental qualification is required for these equipment items for the accident environments to which they may be exposed. These transmitters are part of the reactor protection system and could be exposed to a steam environment consisting of a temperature excursion to 213°F.

D.3 SOLENOID VALVES LOCATED IN THE STEAM AND FEEDLINE PENETRATION AREA

Equipment Item No. 20: Solenoid Valves Actuating Main Steam Isolation Valves
Equipment Item No. 23A: Solenoid Valves Actuating Isolation Valves for the Steam Supply Line to the AFW Pump Turbine (SOV-1310A, B)
Equipment Item No. 23B: Solenoid Valves Actuating Main Feedwater Regulator Valve (PCV-417, 427, 437, 447)

LICENSEE POSITION:

The Licensee states that the pressure and temperature changes for items 20 and 23B are negligible in this area. The Licensee further states that the valves will close (safe position) for all potential modes of failure and that the main steam isolation valves controlled by Item 20 are maintained closed by seat differential pressure.

FRC EVALUATION:

Equipment Item No. 20: The solenoid valves for operating the main steam isolation valves (MSIVs) are located in the steam and feedline penetrations area. These valves operate in a main steam line break (MSLB) accident in order to shut the MSIVs and isolate the steam generators from downstream breaks, or to back up the steam line check valves in preventing the intact steam generators from blowing down through an upstream break.

Equipment Item No. 23A: The solenoid valves for operating the isolation valves for the steam supply line to the auxiliary feedwater pump turbine are also located in the penetrations area. The functioning of these valves is required either to maintain the auxiliary steam line isolation valves open in order to provide steam to the AFW pump turbine following an MSLB or to permit the isolation valves to be closed in case of a break in the auxiliary steam line itself.

Equipment Item No. 23B: The solenoid valves to the feedwater regulator valves isolate the feedwater line in case of a main feedwater line break upon receipt of a safety injection signal. The shut position is not always the safe position for the feedwater regulator valves: on a reactor trip, for example, these valves are initially opened fully to provide additional cooling water. For any accident in which the solenoid valves will be exposed to an abnormal environment, however, the shut position is the safe position.

As noted in Section 4.1.2 of this report, FRC does not agree with the Licensee's claim that the temperature change in this area is negligible. Although the desired position of the valves is shut for the accidents cited, it may be desirable or necessary to reopen one or more of the valves subsequent to the accident to provide for long-term cooling.

FRC CONCLUSION:

The solenoids for operating the MSIVs (Item 20) should be qualified for the environment to which they are subject. They may not be required to function subsequent to their initial closure, provided (i) that the power-operated relief valves are fully qualified and available for discharging steam to atmosphere in order to remove decay heat so that the MSIVs do not have to be reopened to dump steam to the condenser. The solenoids for the feedwater regulators (Item 23B) are required to be qualified. The solenoids for operating SOV-1310A and B (Item 23A) also should be qualified for the environment to which they will be exposed following an MSLB or MFLB in the penetrations area.

D.4 MOTOR OPERATED CONTROL AND ISOLATION VALVES INSIDE CONTAINMENT

Equipment Item No. 7: Accumulator Discharge Valves (MOV-894A,B,C,D)
Equipment Item No. 6: RHR Flow Control Valves (MOV-638, 640)

LICENSEE POSITION:

The Licensee has stated that the accumulator discharge valves (MOV-894A,B,C,D) are not required to be operated after the injection phase and that they are shut upon completion of the injection.

The Licensee has also stated that the RHR flow control valves, MOV-638 and 639 are not normally used during an accident. If adjustment for flow is required, it will be done immediately following switchover to recirculation. Failure does not cause the valve to change position.

FRC EVALUATION:

The accumulator discharge valves (MOV-894A,B,C,D) are normally-open motor-operated gate valves. These valves are checked-open by the safety injection signal at the start of the accident. Accumulators are installed to reflood the core following a design basis accident during the initial blowdown while the safety injection pumps are being started and attaining rated capacity. Accumulator injection begins within seconds of the start of the accident, and the dead-band for starting the active safety injection equipment is generally approximately 30 seconds. Once the accumulators have discharged, the discharge valves are shut as a backup to the check valves, which prevent back-flooding of the accumulators. Since there are two check valves in each accumulator discharge line, the proper operation of these valves following the injection phase of an accident is of little consequence even if the valves are not promptly shut.

The RHR flow control valves (MOV-638 and 640) control flow from the discharge of the RHR heat exchangers directly back to the cold legs of the reactor coolant loops. The Licensee states that these valves are not normally repositioned during the course of an accident except for possible flow adjustments when shifting from the injection phase to the recirculation phase. However, these valves provide flow control for the normal long-term cooling paths from either the containment recirculation pumps or the RHR pumps. FRC does not concur that their continued operation is not required for the long term. The Licensee is committed to installing qualified replacements. There are alternative methods for injecting long-term cooling water into the reactor coolant system (through the RHR hot leg connection or through the safety injection pumps by opening valves MOV 1869A and B) during the interim period until these valves are replaced.

FRC CONCLUSION:

The accumulator discharge valves (Item 7) do not require environmental qualification beyond their short-term function. The RHR flow control valves (Item 6) should be qualified. Alternate methods for injecting cooling water are available until these valves are replaced.

D.5 VALVE POSITION LIMIT SWITCHES ON CONTAINMENT VENTILATION PURGE SUPPLY AND EXHAUST VALVES

Equipment Item No. 28A: Supply Valves (FCV-1170, 1172)
Equipment Item No. 31: Exhaust Valves (FCV-1171, 1173)

LICENSEE POSITION:

Power is administratively removed from [the valve] circuits. Valves are not used while at power. Failure of the limit switch will not cause the valve to change position.

FRC EVALUATION:

The containment purge system at Indian Point Unit 3 is independent of the primary auxiliary building exhaust system and includes provisions for both supply and exhaust air. The supply system includes roughing filters, heating coils, fan, and supply penetration with two butterfly valves for tight shutoff. The exhaust system includes the exhaust penetration with two butterfly valves identical to the supply valves, filter bank with roughing and HEPA filters, fans, and vent. Valves FCV-1170 and FCV-1172 are located inside containment, while valves FCV-1171 and 1173 are located in the piping penetration area. All four butterfly valves perform as containment isolation valves, and they are closed during power operation. The valve position limit switches serve the post-accident function of indicating actual or potential breaching of the barriers to fission product release.

The Licensee has stated that power is administratively removed from the valve actuator and the valves are shut and not used. The implication is that these valves are the same as manually closed containment isolation valves. Since these valves are shut and then de-energized, position indication is not required to verify containment isolation.

FRC CONCLUSION:

FRC concurs with the Licensee's position that containment purge valve position indication need not be environmentally qualified provided the Licensee verifies that appropriate technical specifications and/or procedures preclude opening of these valves during reactor operation.

D.6 VALVE POSITION LIMIT SWITCHES ON CONTAINMENT PRESSURE RELIEF VALVES

Equipment Item No. 28A: (PCV-1190)
Equipment Item No. 28B: (PCV-1191, 1192)

LICENSEE POSITION:

Limit switch is for position indication only, valves are closed on SI and/or containment isolation signal. Once the valve is closed, there is no known failure that would cause the valve to open.

FRC EVALUATION:

The normal pressure changes in the containment during reactor power operation, and during plant cooldown if the containment purge system is not operating, will be accommodated by the containment pressure relief system. This system consists of a pressure relief line equipped with three quick-closing butterfly-type isolation valves, one inside (PCV-1190) and two outside the containment. The valves are automatically actuated to the closed position by safety injection or containment isolation signals.

The pressure relief line presents a direct path from the containment atmosphere to the environment. Containment isolation valve position indication associated with the containment pressure relief system serves the post-accident function of indicating the actual or potential breaching of the barriers to fission product release.

General Design Criterion 55 stipulates acceptable configurations of containment isolation valves. One acceptable combination is one automatic isolation valve inside and one automatic isolation valve outside containment. This notwithstanding, the Licensee has provided one automatic isolation valve inside and two automatic isolation valves outside containment. Therefore, the position indication of all three valves serves a containment isolation function.

FRC CONCLUSION:

The position indication of these valves should be qualified for the environment to which they are subject. The Licensee's position does not eliminate the operator's need to know that the valves are shut and performing their containment isolation function.

D.7 LIMIT SWITCHES IN THE STEAM AND FEEDLINE PENETRATION AREA

Equipment Item No. 30: MSIV Limit Switches

Equipment Item No. 31: Limit Switches for Steam Generator Blowdown Isolation Valves
- Limit Switches for RCS Sampling Isolation Valves
- Limit Switches for Letdown Isolation Valves
- Limit Switches for Excess Letdown Heat Exchanger Isolation Valves
- Limit Switches for Pressurizer Relief Tank Make-up Isolation Valves
- Limit Switches for Pressurizer Steam Space Sample Isolation Valves
- Limit Switches for Containment Sump Discharge Valves
- Limit Switches for Reactor Coolant Drain Tank Vent
- Limit Switches for Pressurizer Liquid Space Sample Isolation Valves
- Limit Switches for Pressurizer Relief Tank Gas Analyzer Isolation Valves
- Limit Switches for Reactor Coolant Drain Tank Gas Analyzer Isolation Valves
- Limit Switches for Reactor Coolant Drain Tank Discharge Valves
- Limit Switches for Steam Generator Sample Isolation Valves
- Limit Switches for Containment Radiation Monitoring Isolation
- Limit Switches for Accumulator Sample Line Isolation Valves
- Limit Switches for Instrument Air Isolation Valve
- Limit Switches for Fan Cooler Service Water Return Valves

Equipment Item No. 32A: Limit Switches for Isolation Valves in the Steam Supply to the AFW Pump Turbine

LICENSEE POSITION:

The Licensee has stated that these limit switches are for position indication only. The Licensee has also stated that, with the exception of the switches for isolating steam to the AFW pump turbine and to the fan cooler service water return valves, the valves are closed on SI and/or containment isolation signal. Once the valve is closed, there is no known failure that would cause the valve to open.

In the case of the MSIVs, the Licensee further indicates that pressure and temperature remain at approximately ambient levels.

FRC EVALUATION:

Although these limit switches provide position indication only, the function of the switches is basically to indicate the proper shutting of the containment isolation valves (with certain exceptions). The closing of containment isolation valves upon receipt of a containment isolation signal requires reliable indication in order for the operator to know that the valves have performed their isolation function. This is particularly true of the MSIVs following a main steam line break accident when the position of the MSIVs may be critical to mitigating the accident and preventing complications with RCS pressure and volume control. Although the valve position information is most important at the start of an accident when many valve operations are being performed, continued reliable position indication is also significant for the long term to prevent possible misinterpretation of valve status by the operators that could result in undesirable operator action.

The isolation valves in the steam supply to the AFW pump turbine are not containment isolation valves; however, they provide a critical function in that they limit the severity of the environment in the auxiliary pump room following a high energy line break to the steam supply and thereby protect a large amount of safety-related equipment. Consequently, the indication that these valves have performed their function is of considerable significance. However, it is noted that the environment is not harsh when the valves and limit switches are required to function.

FRC CONCLUSION:

These valve position indication switches should be qualified for the environment in which the valves perform their isolation function.

D.8 LIMIT SWITCHES IN THE PIPE PENETRATION AREA

Equipment Item No. 32B: Hydrogen Recombiner Containment Isolation Valves IV-2A, 2B, 3A, 5A, and 5B

LICENSEE POSITION:

Limit switches are for position indication only.

FRC EVALUATION:

These limit switches provide indication that the valves have closed upon receipt of a containment isolation signal. This information is significant in the mitigation of accidents in that the operators need indication as to whether or not containment isolation valves have performed their containment isolation function.

FRC CONCLUSION:

Since these valves are installed as containment isolation valves, the limit switches should be qualified for the environment in which the valves perform their containment isolation function.

D.9 TEMPERATURE DETECTORS INSIDE CONTAINMENT

Equipment Item No. 43: Resistance Temperature Detector Elements 420 A and B through 443 A and B (total of 24)

LICENSEE POSITION:

The Licensee has indicated that these detectors are required to function for 5 minutes following an accident.

FRC EVALUATION:

Reactor coolant system temperature indication is required during the initial phases of a design basis accident, during a cooldown to cold shutdown conditions, and during subsequent long-term cooling. The hot-leg detectors aid in determining reactor system subcooling and in providing indication of natural circulation. The cold-leg instruments also provide indication of natural circulation, provide input to heat balance calculations, and provide direct indication of ECCS injection. During plant cooldown, these detectors are necessary to ensure that cooldown rates are not being exceeded. They are also required to ensure that the long-term cooling method is functioning properly.

FRC CONCLUSION:

These temperature detectors should be environmentally qualified because reactor coolant system temperature indication is required to function throughout all phases of accident mitigation, including subsequent long-term cooling.

D.10 ELECTRO-PNEUMATIC PRESSURE TRANSDUCERS LOCATED IN THE PIPE PENETRATIONS AREA

Equipment Item No. 37C: Fan Cooler Service Water Return Valves (TCV-1104 and 1105)

LICENSEE POSITION:

The Licensee states that a review of drawings for these transducers indicates that there is no material that would be substantially affected by the radiation level. The valves are normally open, and air is removed from the controller and the valve following SI initiation. Therefore, there is no known failure mode to position the valve in the unsafe position.

FRC EVALUATION:

In addition to the justification provided by the Licensee for exempting these transducers from qualification, the Indian Point Unit 3 FSAR indicates that there are two separate return paths for the discharge of service water from the fan coil units, only one of which is required for adequate system operation. The FSAR further states that the fan coil units comprise one of two completely independent, 100% capacity containment heat removal systems, the other being the containment spray system.

FRC CONCLUSION:

Environmental qualification of this equipment is not required in view of (i) the Licensee's statement that there is no known unsafe failure mode and (ii) the availability of significant containment heat removal capability to back up the service water discharge lines.

D.11 CONTAINMENT SUMP LEVEL

Equipment Item No. 10: Sump Level Transmitters
(LT-938, 939, 940, 941)

LICENSEE POSITION:

"Two containment and two recirculation sump level instruments are used to monitor level of water in containment during a loss of coolant accident. The instruments are manufactured by DeLaval (Model LS-1900S), and are designed for submerged service at 295°F/60 psig. The primary function of the sump level instrumentation is to ensure adequate water inventory to the suction of the recirculation pumps and containment sump pumps. This can be achieved via the refueling water level instrumentation and equating the volume of water injected to a water level in containment."

FRC EVALUATION:

The sequence of events during the accident at Three Mile Island indicated that the free liquid inventory in the containment building was critical information in the diagnosis of the accident. During the accident, reactor coolant drain tank quench water and primary coolant water vented through the drain tank relief valve and flowed to the reactor building sump. Water within the containment sump was then discharged to the auxiliary building sump tank and thus resulted in some transfer of radioactive material outside of the containment building. The accumulation of water in the TMI-2 containment may have contributed to equipment failure due to flooding.

Containment sump water level instrumentation provides indication of leakage within containment and of adequate water inventory for performance of the ECCS. The containment sump water level instrumentation serves the post-accident function of providing information to monitor the process of accomplishing critical safety functions. As a consequence, the NRC has included containment water level monitors in the TMI Lessons Learned instrumentation requirements for short-term action as recommended by the Advisory Committee on Reactor Safeguards. Specifically, a continuous indication of containment water level is to be provided in the control room. A narrow-range instrument is to be provided to cover the range from the bottom to the top of the containment sump. In addition, a wide-range instrument is to be provided to cover the range from the bottom of the containment to the elevation equivalent of a 600,000-gallon capacity.

FRC CONCLUSION:

The containment sump level instrumentation should be qualified for the environment to which it is subject. The Licensee's position that the refueling water level instrumentation can be used to determine the volume of water injected and the containment water level relies on additional unqualified equipment and operator actions.

DELETED MATERIAL IS PROPRIETARY INFORMATION

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

EQUIPMENT ITEM NO.	DRAFT INTERIM TECHNICAL EVALUATION REPORT SECTION	FINAL TECHNICAL EVALUATION REPORT SECTION
1	3.3.2.1	4.3.3.1
2	3.3.2.1	4.3.3.1
3	3.3.2.1	4.3.3.1
4A	3.2.1	4.3.3.2
4B	3.2.1	4.3.3.2
5A	3.3.1.1	4.3.3.3
5B	3.3.1.1	4.3.3.3
6	3.3.2.2	4.5.2.1
7	3.3.2.2	4.3.3.4
8	3.3.1.1	4.3.3.3
9	3.3.1.1	4.3.3.3
10	3.3.3.1	4.6.1
11A	3.3.2.3	4.5.2.2
11B	3.3.2.3	4.5.2.2
12	3.3.2.3	4.5.2.2
13	3.3.2.6	4.5.2.5
14A	3.3.2.4	4.5.2.3
14B	3.3.2.5	4.5.2.4
14C	3.3.2.5	4.5.2.4
15	3.3.2.4	4.5.2.3
16A	3.3.2.7	4.5.2.6
16B	3.3.2.3	4.5.2.2
17	3.3.2.8	4.6.2
18A	3.3.2.9	4.3.1.1
18B	3.3.2.9	4.6.3
19	3.3.2.11	4.6.9
20	3.3.2.12	4.6.11
21	3.3.2.10	4.5.2.7
22	3.3.2.10	4.5.2.7
23A	3.3.2.12	4.6.11
23B	3.3.2.11	4.5.2.12
24	3.3.2.10	4.5.2.7
25	3.3.2.10	4.5.2.7
26	3.3.1.2	4.6.12
27	3.3.2.13	4.3.2.2
28A	3.3.1.3	4.3.1.2
28B	3.3.1.3	4.3.1.2
29	3.3.2.14	4.6.4
30	3.3.1.4	4.6.10

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

EQUIPMENT ITEM NO.	DRAFT INTERIM TECHNICAL EVALUATION REPORT SECTION	FINAL TECHNICAL EVALUATION REPORT SECTION
31	3.3.1.5	4.6.10
32A	3.3.1.6	4.7.1
32B	3.3.1.7	4.3.2.3
33	3.3.1.8	4.3.1.3
34A	3.3.2.15	4.6.5
34B	3.3.2.16	4.7.2
34C	3.3.2.16	4.7.2
35	3.3.2.17	4.5.2.8
36	3.3.2.17	4.5.2.8
37A	3.3.2.18	4.3.1.5
37B	3.3.2.18	4.3.1.5
37C	3.3.1.10	4.4.1
38	3.3.2.19	4.6.6
39	3.3.2.20	4.6.7
40A	3.3.2.21	4.5.2.9
40B	3.3.2.21	4.5.2.9
41	3.3.2.22	4.5.2.10
42A	3.3.2.23	4.3.2.1
42B	3.3.2.24	4.6.8
43	3.3.2.25	4.5.2.11
44	--	4.3.1.4
45	3.3.1.5	4.4.2

DELETED MATERIAL IS PROPRIETARY INFORMATION

APPENDIX F - PROPERTIES OF CAST PHENOLIC RESINS

PHYSICAL PROPERTIES

	Specific Gravity	Specific Heat	Thermal Conductivity (c.g.s. units) $\times 10^{-4}$	Thermal Expansion Coeffs. (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	0.35-0.36	4-12	3-6	70-150
Chopped-cotton-fabric-filled	1.3-1.4	0.30-0.35	3-5	2-6	200-400
Mineral-filled	1.6-2.4	0.25-0.35	8-20	2-4	20-100
Laminated Material					
Paper-filled	1.3-1.4	0.3-0.4	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

MECHANICAL PROPERTIES

	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	3-15	3-10	15-40	1,000-1,500	300-500	0.1-0.5
Chopped-cotton-fabric-filled	5-8	3-15	10-15	20-35	700-1,200	300-500	0.3-3.0
Mineral-filled	4-8	3-15	4-15	20-35	1,000-2,500		0.1-1.0
Laminated Material							
Paper-filled	3-25	15-30	5-12	20-40	1,000-3,000		0.2-2.0
Fabric-filled	3-20	15-30	5-12	30-45	500-1,500		1-5
Asbestos-filled	7-12	10-15	4-8	30-50	500-2,000		0.2-1.0

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

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

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

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

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

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

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

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

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

During LOCA, the dose rates calculated in accordance with conservative NRC recommendations are typically 1 to 3 Mrd/h gamma and 10 Mrd/h beta during the first 10 hours of the LOCA. (These data are for a nominal 1000 MW(e) plant.)

It can be seen that the dose rates for insulation subject to beta radiation exceed most test radiation dose rates by an order of magnitude.

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

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