TECHNICAL EVALUATION REPORT

EQUIPMENT ENVIRONMENTAL QUALIFICATION

DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR

NRC DOCKET NO. 50-409

NRC TAC NO. 42522

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC TASK 203

Prepared by

Franklin Research Center
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FRC Group Leader: C. J. Crane

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

NRC Lead Engineer: J. Lombardo

April 15, 1981

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April 15, 1981

United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Edward J. Butcher, Jr.
Project Officer (MS 416)

Reference:	Utility:	Dairyland Power Cooperative
	Plant Name:	La Crosse Boiling Water Reactor
	Docket No.:	50-409
	NRC TAC No.:	42522
	NRC Contract No.:	NRC-03-79-118
	FRC Project No.:	C5257
	FRC Task No.:	203
	Project:	SEP Plants Electrical Equipment Environmental Qualification (EEQ)

Subject: Final Technical Evaluation Report - Formal Issue
Equipment Environmental Qualification

Dear Mr. Butcher:

Enclosed please find the Final Technical Evaluation Report on the referenced subject.

Very truly yours,

S P Carfagno

S. P. Carfagno
Project Manager

SPC/CJC/cm

Enclosure

cc: Z. Rosztoczy - Letter only
J. Lombardo - 5 Reports
P. DiBenedetto - Letter only

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Subcontractors:

ECI = Energy Consultants, Inc.

SRC = Synergic Resources Corporation

W = WESTEC Services, Inc.

1. INTRODUCTION

1.1 PURPOSE OF THE EVALUATION

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

1.2 GENERIC ISSUE BACKGROUND

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

Qualification criteria applied during the licensing of 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, 383-74, 317-76, 334-74, 381-77, 382-80, and 627-80. NRC NUREG documents 0413 and 0588 have been developed to address this topic. In particular, NUREG-0588 (published for comment in December 1979) formally presented the NRC staff positions regarding selected areas of environmental qualification of safety-related electrical equipment in the resolution of General Technical Activity A-24, "Qualification of Class IE Safety Related Equipment." The positions documented therein are applicable 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" [3].* (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 present the results in the form of a Technical Evaluation Report for the 11 oldest plants (included in the SEP review).

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

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" [3], 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 [27], specifying that licensees and applicants must meet the requirements set forth in the DOR Guidelines and NUREG-0588 regarding environmental qualification of safety-related electrical equipment in order to satisfy 10CFR50, Appendix A, General Design Criteria, Section I, Criterion 4. This Order also established that the Safety Evaluation Reports on this subject, to be prepared by the NRC staff, must be issued on February 1, 1981 and that all subsequent actions to be taken by licensees to achieve full compliance with the DOR Guidelines or NUREG-0588 must be completed no later than June 30, 1982.

1.3 SPECIFIC ISSUE BACKGROUND

In a letter dated December 23, 1977, the Dairyland Power Cooperative (DPC) was formally notified by the NRC that the review of environmental qualification for safety-related equipment for the LaCrosse Boiling Water Reactor (LACBWR) would be conducted under SEP Topic III-12. Information requested from DPC included identification of electrical equipment required to perform safety functions while subject to design basis accident environments, definitions of environmental service conditions at equipment locations, and the status of environmental qualification of equipment and identification of supporting documentation. On January 30, 1978, the NRC conducted a plant visit to LACBWR to explain the purpose of the environmental qualification review program and to provide guidance with regard to the contents of the Licensee's response to the December 23, 1977 letter. In response to the NRC request, DPC provided information via submittal letters dated February 22 [7] and October 26, 1978 [6], and April 26 [4] and May 29, 1979 [5].

On February 15, 1980, NRC qualification guidelines for identification and evaluation of safety-related equipment [3] were transmitted to DPC.

In March 1980, the NRC transmitted to DPC schedule information [1,2] relative to the SEP Environmental Equipment Qualification Program.

During the week of September 7, 1980, NRC and FRC representatives visited the LACBWR site, inspected safety-related systems and components, identified and tabulated safety-related systems and components by discussions with plant personnel, and conducted a general overview of the DPC submittals on environmental qualification. At this meeting, qualification documentation, supplemental information, and manufacturers' information were provided to NRC and FRC.

During the weeks of September 14, 21, and 29, 1980, additional information and clarification were provided by DPC to FRC and NRC for use in evaluation of previous submittals.

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

FRC issued a Draft Interim Technical Evaluation Report (DITER) to the NRC on October 13, 1980 [28]. Copies of the report were transmitted to DPC by the NRC.

On October 31, 1980, additional responses and qualification information relative to the DITER were provided by the Licensee [23].

1.4 SCOPE OF THE EVALUATION

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

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

2. NRC CRITERIA FOR ENVIRONMENTAL QUALIFICATION

2.1 CRITERIA PROVIDED BY THE NRC

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

- "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" [3]
- "Guidelines for Identification of That Safety Equipment of SEP Operating Reactors for Which Environmental Qualification Is To Be Addressed" [3].

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 [25], "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 conditions 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 [25] 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 [25] permits deferment of the review of environmental qualification for all safety-related equipment items located in plant areas where the equipment is not exposed to the direct effects of a HELB or to nuclear radiation emanating from circulation of fluids containing radioactive substances. At the licensee's option, the review may be deferred until after February 1, 1981.

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

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

The Guidelines require that the test chamber environment envelop the required service conditions for a time equal to the period from the initiation of the accident until the service conditions return to normal. Supplement 2 to IE Bulletin 79-01B [25] 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 [26] permits the submittal of qualification documentation regarding the TMI Action Plan equipment and the equipment required to achieve and maintain a cold shutdown condition to be delayed as follows:

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

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

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

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

- The qualification information for equipment required to achieve and maintain a Cold Shutdown condition ... will be submitted not later than February 1, 1981."

2.2.6 TEST SEQUENCE
(DOR Guidelines Section 5.2.3)

Supplement 2 to IE Bulletin 79-01B [25] 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 [25] 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, Dairyland Power Cooperative, identified a relatively small number (31) of safety-related electrical equipment* items located in various areas of the LaCrosse Boiling Water Reactor in its submittals to the NRC [4-7,23]. 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 Transmitters, Fischer & Porter, Model 10B2496, located within containment). Appendix A describes the environmental service conditions for each location, Appendix B tabulates the equipment items and locations (the tabulation does not include equipment covered by the evaluation deferment described in Section 2.2.3 of this report), and Appendix C lists the plant systems identified by the Licensee and the NRC as being essential to safety.

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

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

Topics not within the scope of FRC evaluation are:

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

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*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 October 31, 1980 [28]. 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 [67,23]. This information was analyzed by FRC to determine:

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

All information was reviewed by FRC for conformance to the NRC criteria referenced in Section 2 of this report. As requested by the NRC, FRC reviewers used all qualification information developed in the Equipment Environmental Qualification (EEQ) program, 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 where appropriate. Based on the FRC conclusion, each equipment item was assigned to one of the generic qualification categories provided by the NRC. The NRC category descriptions follow.

NRC CATEGORIES AND DEFINITIONS

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

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

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

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

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

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

- NRC Category II.b
  EQUIPMENT THAT SATISFIES ALL APPLICABLE REQUIREMENTS OF THE DOR GUIDELINES WITH THE EXCEPTION OF QUALIFIED LIFE 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 the qualified life criterion. With respect to qualified life, the equipment items have been found to have a qualified life which (1) is limited to a time interval less than plant life, (2) has not been adequately established in terms of calendar time, or (3) has not been evaluated by the licensee.

- NRC Category II.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.

o NRC Category V
EQUIPMENT THAT IS UNQUALIFIED

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

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

o 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 obs :vations 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 items are defined in Appendix B.

4.1 METHODOLOGY USED BY THE LICENSEE

The Licensee, Dairyland Power Cooperative (DPC), has provided only limited information with regard to the methodology used to select safety-related equipment and to develop the other information necessary to comply with the requirements imposed by the NRC Memorandum and Order dated May 23, 1980 [27] and the DOR Guidelines [24]. The Licensee has responded [23] to most of the equipment questions raised in the FRC Draft Interim Technical Evaluation Report (DITER) for the LaCrosse Boiling Water Reactor (LACBWR), but did not resolve the concerns presented in the Conclusion section or in Appendix G.

The DITER identified four items requiring Licensee action. The Licensee has acted on only one of the four: environmental service conditions for HELB

* 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).

areas located outside containment have now been described. These action items are discussed in Appendix F. A review of DPC's final submittal has generated the following observations and concerns.

4.1.1 COMPLETENESS OF EQUIPMENT LIST

The Licensee has identified only 31 specific equipment items as being safety-related and exposed to a harsh environment. The Licensee's submittals [6,7] and responses to the DITER equipment item reviews did not focus on demonstrating equipment qualification, but instead presented systems analyses based on various degrees of engineering judgment which indicated that if the equipment should fail, it would do so in a fail-safe mode. Detailed qualification analyses of individual equipment components that would better assess the likelihood of operability during postulated accidents were not conducted.

The major portion of the Licensee's analyses postulated that other available equipment or systems could perform the required functions if certain equipment items failed. Because the Licensee did not thoroughly demonstrate that all of the backup systems are safety-related and also failed to provide qualification documentation to demonstrate continued performance under accident conditions, some of the Licensee's systems positions did not adequately address the Guidelines requirements. The Licensee's systems positions and FRC's evaluations are discussed in Appendix H of this report.

The concern that not all safety-related electrical equipment exposed to a "harsh" environment had been identified by the Licensee was raised in the Conclusion section of the DITER. Six specific equipment items were listed by FRC as examples of possibly omitted equipment:

- o overhead storage tank level indicator
- o reactor control rod drive scram solenoids
- o reactor or main steam pressure transmitters
- o main steam flow transmitters
- o reactor protection system instrumentation
- o safety-related control stations.

In Reference 23, the Licensee contends that none of these items, except for some reactor protection system instruments, requires qualification. FRC disagrees with this contention in most cases and presents its evaluation and the Licensee's statement in Appendix I.

In addition to reconsidering the six items listed above, the Licensee should confirm the locations of both the safety-related reactor scram system switchgear and the switchgear associated with all safety-related pumps, including those which supply cooling water to safety-related HVAC systems.

A significant concern expressed in both the Conclusion section and Appendix G of the DITER refers to the Licensee's failure to demonstrate conclusively that the LACBWR has a viable safety-related post-accident heat dissipation system for the containment and the reactor. The Licensee has provided brief calculations of the capacity of a proposed component-cooling heat removal system to cool the containment, but did not include any detailed documentation such as drawings and overall heat transfer calculations. Moreover, calculations were not provided to demonstrate that the component-cooling system could also serve as the reactor vessel's long-term heat sink.

The Licensee failed to respond to the DOR Guidelines requirement for maintenance surveillance of safety-related equipment subject to age-related degradation. The Licensee should presently be reviewing maintenance records to determine if the qualified life of equipment is affected. Such a surveillance program could indicate the need for replacement of equipment items on a more frequent basis.

4.1.2 ENVIRONMENTAL SERVICE CONDITIONS

The Licensee's October 31, 1980 submittal provided additional information on the pressure and temperatures anticipated in the turbine building during a MSLB or HELB. Expected duration of high humidity conditions was not discussed. The profile curves, which were not referenced to a specific study, indicate that the temperature and pressure rise but return to near-normal conditions in a few minutes. It should be noted that detailed calculations supporting the Licensee's conclusions have been lacking in all of the data submittals.

A calculation performed by the Licensee showed that the volume of water flooding the pipe tunnel from a main feedwater pipe break was expected to be 3,283 cubic feet, but indicated neither the maximum water level nor the equipment likely to be affected. This environmental service condition will require further review and analysis by the Licensee.

4.1.3 AGING AND QUALIFIED LIFE

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

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

Qualified life is the maximum period of normal service, under specified conditions, for which it can be demonstrated that, at the end of the period, the equipment is still able 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 components 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 31, with the possible exception of certain simple materials, there is no rigorous basis for establishing equipment qualified lifetimes 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.

The Licensee should review the qualified life values and the present installed life of the safety-related 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 that are fully acceptable on the basis that (1) all criteria defined in Section 2 of this report are satisfied or (2) sufficient data exist to determine that specific deviations are acceptable.

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

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

4.2.1.1 Equipment Item No. 17
Electrical Penetrations Located Inside Containment
Special Design for LaCrosse
(Licensee References TR-8 and TR-9)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

Licensee References TR-8 and TR-9 are analyses of the integrity of the containment building penetration and qualification of electrical cable containment penetrations to withstand a LOCA. In TR-8, the Licensee describes the effects of electrical faults, their interaction with containment electrical penetrations, operator actions to mitigate the effects of faults, and modifications involving addition of fuses to circuits for seal injection pumps 1A and 1B, shield cooling pump, and CRD nozzle pumps. In TR-9, the Licensee describes data and tests on the various elements of the penetrations, i.e., metal parts, MI cable, mechanical compression fitting, UL tests on epoxy seals, and flame tests conducted by LaCrosse on the penetrations.

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

a. The Guidelines require that complete and auditable records reflecting a comprehensive methodology be referenced and available for review of all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E equipment located inside containment required to mitigate the consequences of a design basis event. A design specification is not sufficient. Qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. The type test is valid only if the test specimen and installed equipment have the same design, material, and production procedures. An analysis of the impact of deviations between the test specimen and installed equipment is an essential part of the qualification documentation.

b. FRC agrees that the individual mechanical and metal parts would not be significantly affected by the LaCrosse LOCA conditions. However, as noted in 3.3.2.3, there are some epoxies that are adversely affected by humidity, radiation, and temperature combined. It is suggested that the Licensee obtain information from the manufacturer on the performance of the epoxy used for end sealing the MI cable penetration under combined radiation, humidity, and thermal aging.

c. An inspection of the drawing identified as Fig. 2.3.4-4 in Reference TR-9 indicates that under LOCA conditions there would be induced thermal strains in the MI cable sheath, ferrule, and gland threads as a result of differential thermal expansion between the rather massive penetration and the MI cables passing through. Analysis or test of the effects should be provided to confirm the integrity of the penetrating cable and leak tightness of the assembly.

LICENSEE RESPONSE:

Mineral Insulated Cable Penetrations

The epoxy sealant used in most terminations of mineral insulated cable has previously been addressed (Item 33 of Enclosure 2). The thermal strain present under LOCA conditions of the containment penetration has been analyzed and found to be of a low enough magnitude to have no effect on penetration integrity (see Enclosure 5).

FRC EVALUATION:

The Licensee has provided additional documentation, "LOCA Environmental Effects on Containment Mineral Insulated Cable Penetrations," which analyzes the potential impact on the electrical penetration assembly of the rapid heating immediately following a LOCA. The analysis deals with the materials expected to experience the greatest thermal expansion and with the possibility of resulting leakage. The Licensee states that thermal expansion during the rapid heat-up stage of a LOCA would be expected to reduce any possible steam leak paths. Concerning thermal contraction during cooldown, the analysis anticipates no leakage increases because the cooldown rate is expected to be significantly slower than the heat-up rate. The supporting calculations and detailed drawings should be made a permanent part of the Licensee's EEQ file.

The Licensee has maintained that the penetration assembly shields the epoxy sealant from LOCA conditions. This shielding effect was verified by FRC observation during a site visit and also applies to MSLB conditions.

FRC CONCLUSION:

This equipment is assigned to NRC Category I.a because no materials which can degrade significantly are subject to harsh environmental conditions.

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

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

For the LaCrosse Boiling Water Reactor, no equipment falls within this category.

4.3 EQUIPMENT QUALIFIED WITH RESTRICTIONS

This section includes equipment items that are acceptable on the basis that (1) all criteria defined in Section 2 of this report are satisfied; (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.

For the LaCrosse Boiling Water Reactor, no equipment falls within this category.

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

The equipment items in this section will be acceptable and will satisfy all applicable criteria defined in the DOR Guidelines with the exception of qualified life provided that specific modifications are made on or before the designated date. When the modifications are complete, the equipment can be considered qualified with the exception of the qualified life criterion. With respect to qualified life, the equipment items have been found to have a qualified life which (1) is limited to a time interval less than plant life,

(2) has not been adequately established in terms of calendar time, or (3) has not been evaluated by the Licensee.

For the La Crosse Boiling Water Reactor, no equipment falls within this category.

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

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

4.3.3.1 Equipment Item No. 11
Mineral Insulated Cable Located Inside Containment
Manufacturer and Model Not Stated
(Licensee References 9 and 17)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee states that specific testing on mineral insulated (MI) cable has not been conducted and notes that testing is not considered necessary because the materials are not subject to degradation by the LOCA environment as required by the DOR Guidelines. It is further stated that the ends are sealed by epoxy and that the seal will not track (seep) beyond 4 to 5 inches should it leak moisture.

FRC has reviewed the Licensee submittal and agrees that the MI cable does not contain materials that would be adversely affected by the LOCA conditions and therefore does not require qualification testing. However, tests have been conducted on various epoxy formulations; some formulations are adversely affected by high humidity (softening and breakdown) and some are affected by radiation, humidity, and temperature combined. These data are contained in proprietary reports.

It is suggested that the Licensee obtain information from the manufacturer on the performance of the epoxy used for sealing the MI cable under combined radiation, humidity, and thermal aging.

LICENSEE RESPONSE:

Mineral Insulated Cable Epoxy (Item 33 of Enclosure 2)

The epoxy was used in terminal boxes on the ends of mineral insulated cable. All safety-related MI cable terminations are located inside water-tight housings. No safety-related MI cable terminations are exposed to the LOCA atmosphere. The epoxy, therefore, should not be an item in the qualification program.

FRC EVALUATION:

The Licensee has stated that the mineral insulated cable's epoxy system is used only on the ends of the cables, which are enclosed in watertight housings. Although the Licensee did not submit drawings showing how this technique of enclosing the epoxy prevents its exposure to the harsh LOCA environment, the design of this mineral insulated system with its enclosed end terminations is reasonable, and the system should not degrade except for aging (see Section 4.1.3).

FRC CONCLUSION:

This equipment item is assigned to NRC Category II.c because radiation and temperature can degrade the epoxy. The Licensee should establish a conservative qualified life value.

4.3.3.2 Equipment Item No. 12
Limit Switches Located Inside Containment
NAMCO Model EA-180
(Licensee Reference 18)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

The Licensee stated that limit switches for the reactor steam relief valves located in containment are NAMCO Model EA-180, and it provided Reference 18 as evidence of qualification. Reference 18 is a certification data sheet from NAMCO stating that the switches provided to LaCrosse are identical to the design that was tested to meet requirements of IEEE Std 323-74 and reported in QTR-105.

FRC has reviewed QTR-105 and agrees that the limit switch meets the DOR Guidelines and that the test conditions envelop the LOCA data provided in Reference 7. However, the NAMCO test report emphasizes that the sealant used in the actual installation must prevent entry of steam into the switch. Accordingly, the Licensee should establish that the LaCrosse installation uses a sealant that the manufacturer (NAMCO) has determined to be satisfactory.

LICENSEE RESPONSE:

Safety Valve Position Switch Housing Sealant (Item 39 of Enclosure 2)

Our discussion with NAMCO revealed a silicone rubber sealant was used for qualification testing. The sealant used to close the environmental housings on these switches must meet the criteria specified in the qualification test by NAMCO. LACBWR used General Electric RTV-11 silicone rubber sealant.

NAMCO has agreed to furnish LACBWR a written statement on the specific sealant used in their test program and LACBWR will compare its actual material to the manufacturer's. If the material used at LACBWR is not comparable, it will be replaced with an acceptable sealant at the first plant outage of sufficient duration following receipt of the specified material.

FRC EVALUATION:

FRC agrees with the approach taken by the Licensee to replace the existing sealant (GE RTV-11 silicone rubber sealant) if the sealant already installed in the plant is a different type than was tested by NAMCO. It is questionable, however, that the sealant material will not degrade over the life of the plant. The Licensee should provide a qualified life statement for these limit switches based on the expected life of the sealant and other switch materials that may degrade with time (see Section 4.1.3).

FRC CONCLUSION:

The limit switch is assigned to NRC Category II.c because FRC has reviewed test reports which the Licensee did not provide and testing was satisfactory. The Licensee should establish a conservative qualified life value.

4.3.3.3 Equipment Item Nos. 9 and 16
Motorized Valve Actuators Located in Turbine Building
Limitorque, Model Not Stated
Actuates Alternate Core Spray System Valves
(Licensee Reference 13)

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

The Licensee identified safety-related Limitorque motorized valve actuators (MVAs) outside containment that could be exposed to a harsh environment as a result of a steam line break. Reference 13 was cited as evidence of qualification.

FRC has reviewed Reference 13 and the test is judged to envelop the conditions identified for the steam line break in Notes 7 and 8 of Reference 1 and Reference 11 for pipe breaks outside containment. Based on this review, FRC finds that this equipment satisfies the requirements of the DOR Guidelines.

LICENSEE RESPONSE:

[No response provided.]

FRC EVALUATION:

Although Reference 13 adequately envelopes the environmental service conditions, the Licensee has not provided any evidence that this reference is applicable to the installed equipment. Also, a statement regarding qualified life has not been provided (see Section 4.1.3).

The Licensee should contact Limitorque Corporation to obtain certification that the cited reference (or an equivalent one) is applicable to the installed units. FRC expects that the Licensee will be able to establish qualification, because of the extensive amount of testing that has been performed on Limitorque MVAs. The Licensee also should review maintenance records to determine whether any of the MVA's components are experiencing aging-related degradation.

FRC CONCLUSION:

This equipment is assigned to NRC Category II.c. FRC believes that the Licensee will be able to demonstrate that all Guidelines requirements are satisfied. A conservative qualified life value for this equipment should be established by the Licensee.

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 Nos. 19A and 19B
Solenoid Valve and Limit Switches Located Inside Containment
19A: ASCO Model 8300B9RF
19B: Limit Switches, Manufacturer and Model Not Stated
Actuates Reactor Cavity Vent Valve and Provides Position Indication
(Licensee Reference TR-6)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

In the table for the reactor building heating, ventilation, and air conditioning system of Reference TR-6, the Licensee identifies solenoid valves and limit switches for reactor cavity vent application and states that a LOCA causes loss of power, which, in turn, causes the valves to fail in the open position (closed to the ventilator exhaust fan inlet).

Since there is no model or manufacturer listed for the solenoid valve, FRC cannot independently confirm that the valve will fail as indicated. [Note: The model and manufacturer were provided in the Licensee's final submittal.] The Licensee should provide test data or analysis that the valve will fail as identified so that the NRC can evaluate the safety issue discussed in Reference TR-6.

LICENSEE RESPONSE:

> This equipment (control valve 55-25-001 and solenoid valve 55-25-011) should be removed from the list of equipment requiring environmental qualification. This control valve routes the reactor cavity and fuel storage well vent line to either the containment building ventilation system or the 4-inch containment vessel offgas vent header. Both of these routings have downstream containment isolation valves which are covered in the equipment qualification program. This three-way valve is not safety-related nor will it be required to function in a post-LOCA environment.

[illegible]LUATION:

FRC agrees with the Licensee's position.

FRC CONCLUSION:

This equipment is assigned to NRC Category III because it is required for normal plant operation 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.

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

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

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

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

For the LaCrosse Boiling Water Reactor, no equipment falls within this category.

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

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

4.5.2.1 Equipment Item No. 7B
Level Transmitter Located Inside Containment
Foxboro Model E13DM
Reactor Water Level, Water Level No. 3 - Wide Range (50-42-306)
(Licensee Reference TR-11)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

Licensee Reference TR-11 consisted of a manufacturer's installation instruction, which cannot be substituted for a qualification test or analysis report. The Licensee has shrouded the transmitter in individual watertight enclosures for protection against spray, flooding, and pressure. The cable is sealed copper-sheathed mineral insulated for watertight application, and the entire assembly was air pressure tested at 52 psig for several hours. The transmitter's enclosure is gasketed, so long-term aging and susceptibility to eventual leakage has not been demonstrated; therefore, total enclosure qualification is lacking. In addition, the transmitter's amplifier is remotely mounted outside containment; this should provide increased operational reliability even though the transmitter's electric force balance feedback coil component is still inside containment.

The Licensee contends that the transmitter's primary function of emergency core cooling system (ECCS) automatic initiation would be performed prior to significant exposure to the severe LOCA environment. Therefore, its possible loss would mean that ECCS water could not be added in a controlled manner.

FRC has reviewed various test records and has found that the Foxboro Corporation has tested the E13DM transmitter under LOCA conditions; however, qualified life of the unit was not addressed. A review of Westinghouse Report WCAP-8541 is provided below as guidance and information for the Licensee.

a. WCAP-8541 contains descriptions of and results from the following qualification programs conducted for the Foxboro Company by various test organizations:

   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.

   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 used amplifier part numbers N0148ND, N0148PF, and N0148NL.

   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. Failure of certain transmitters at high radiation levels was noted.

   Report No. T3-1097 -- A radiation exposure test was conducted on improved amplifiers, and modified because of the failures experienced during the previous test.

   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. WCAP-8541 presents 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. The Licensee has identified the Foxboro transmitter overall model numbers; however, many specific details concerning transmitter identification have not been presented.

   In order to establish the relationship between the test specimen and the installed equipment, FRC concludes that the Licensee must provide the following additional information for the installed equipment:

- The full model number for all transmitters (for example, E13DM-1SAM2).
- The transmitter case style (for example, A or B).
- The transmitter current output rating (for example, 4 to 20 mA dc or 10 to 50 mA dc).
- The transmitter top works amplifier part number (for example, N0148PW).
- The transmitter body material (for example, aluminum, iron, or stainless steel).
- The transmitter capsule assembly part number and O-ring part number (and material).
- The method of electrical connection and associated accessories (for example, Conax fitting and pressure seal junction box assembly).
- The transmitter special modification designation (for example, MCA/RRW).

c. The second LOCA test program (T3-1013) was more comprehensive than the first (Q9-6005). Various "Style B" transmitters with cast iron covers were tested. Westinghouse has stated that the greater heat sink provided by the cast iron cover should improve test results over the aluminum cover; however, a specific comparison of test results was not presented. FRC concludes that, for the purpose of establishing qualification, the test program reported in T3-1013 can be considered the primary test.

d. The Licensee has stated that these transmitters are totally enclosed in a watertight container and therefore submergence will have no effect on the units. FRC concludes that submergence testing of the transmitters is not required. However, the Licensee should provide evidence of periodic presure testing of the enclosure and an analysis of aging degradation of gaskets and seals.

e. The Guidelines require that the test chamber temperature/pressure profile envelop the service conditions for a time duration equivalent to the period from the initiation of the accident until the service conditions return to normal values. Test Report No. T3-1013 has established that the test chamber temperature/pressure profile under all steam conditions, including chemical spray, exceeded the postulated accident profile (with the exception of time duration as discussed in Appendix A); therefore, FRC concludes that this aspect of the qualification program is acceptable.

f. The Guidelines require that equipment operational modes during testing should be representative of the actual plant application requirements. In addition, failure criteria should include instrument accuracy requirements.

Test Report No. T3-1013 states that the output error for the E13DM-1SAH2 transmitter ranged from +1.6% to -2.0% during the initial 90 minutes and settled at -1.5% for the remainder of the test. The output error for the E13DH-1SAM2 ranged from +4.0% to -0.05% during the initial 90 minutes and settled at +0.5% for the remainder of the test. This is presumed 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 Company description of the test states that 3-XJB-I/25 MCA cast iron junction boxes and pressure seal assemblies (including NO148PQ terminal blocks) were tested; however, no reference was made to Conax. The assembly performance was satisfactory. 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 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: E13DM-1SAM2 (3 specimens) using 4-20 mA dc N0148ND and N0148NL amplifiers, E13EH-1SAM2 (3 specimens) using 4-20 mA dc N0148NL amplifiers, and E13DM-HSAM2 (2 specimens) using 10-50 mA dc N0148PD amplifiers. Two of the specimens were previously tested (T3-1013) in a steam-air chemical spray environment. These units were designated serial number 2692438 (Model E13DM-1SAM2 using a 4-20 mA dc N0148ND amplifier) and 2692441 (Model E13DH-1SAM2 using a 4-20 mA dc N0148NL amplifier). It should be noted that amplifier N0148NL was a prototype unit designed for nuclear service with radiation-resistant wiring. Some discrepancies exist in the referenced report; the test organization report states that the amplifier for serial number 2692441 (previously tested) was remote-mounted outside the radiation field, while the Foxboro summary of this report states that only the amplifier for serial number 2692442 transmitter was located remote from the radiation source.

The summary conclusion of the test was that all units continued to function at a dose rate of 1 Mrd/hr to total doses of 76 Mrd or greater. However, failures did occur. The two 10-50 mA dc transmitters, Model E13DM-HSAM2 (using the N0148PD amplifier), failed: one unit's output went to 0% at 76 Mrd and then returned to half the normal output; the other unit's output went to half normal at 90 Mrd. Two of the 4-20 mA dc transmitters, Model E13DH-1SAM2 (using the N0148NL amplifier), continued to function with maximum errors of 3.75% until termination of the test. The unit with serial number 2692441 (previously tested) operated with a maximum error shift of -3.3% up to a failure point of 86 Mrd. The E13DM-1SAM2 transmitters using the N0148ND amplifier operated with maximum error shifts of 4.85% up to 22 Mrd. The other unit (N0148NL amplifier) exhibited possible failure for 2 hours at the 69-Mrd level of irradiation.

Failure of the amplifiers, both the 4-20 and the 10-50 mA models, was traced to a voltage drop across a type 1N4148 diode. This diode is used in all three amplifier models. Although failure occurred at radiation levels greater than the postulated accident levels for gamma radiation (20 Mrd), FRC concludes that degradation due to radiation did occur to units that were not simultaneously or subsequently exposed to a steam-air chemical spray environment. The Guidelines require that radiation exposure should be applied during the test sequence concurrent with, or prior to, the temperature and pressure/steam environment if it is known that the device contains materials that can be degraded by irradiation. It has been established that the transmitters are susceptible to degradation by radiation exposure. 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. However, the LaCrosse units have remote amplifiers located in the control room. The Licensee should analyze the applicability of this test to LaCrosse's specific design.

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 occurred during the previous test program (T3-1068), replaced diode type 1N4148 with type 1N645. In addition, certain resistors and capacitors were replaced in the 10-50 mA dc amplifiers. Up to 22 Mrd, the N0148NL amplifiers exhibited maximum shifts of -6% zero, +1% span, and -4.7% to -5.7% output. The N0148TE(TJ) amplifier exhibited maximum shifts of -2.5% zero, +0.5% span, and -2% output. The N0148PW amplifiers experienced some difficulty. Two units functioned to 220 Mrd, and one unit became erratic at 140 Mrd and then failed. Maximum shifts for the amplifiers were -4.2% output, 4.2% zero, and 2.2% span. The failure was traced to a type 2N1711 transistor, which the report states is being analyzed.

Although the units were tested to radiation levels greater than the postulated accident level, FRC concludes that these specific amplifier assemblies have not been tested as an integral part of transmitters 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 indicated 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 exact relationship between the installed transmitters and the appropriate test specimens has not been established. The Licensee should provide this detailed information (as indicated in Item b).
2. The Licensee should provide detailed information regarding the method of electrical connection at the transmitter for the test specimens and the installed units.
3. The test report indicated that the transmitters are degraded by radiation. The Licensee should provide evidence that radiation testing combined with a LOCA temperature/pressure exposure is not required due to LaCrosse's unique design.
4. The Licensee should address the matter of qualified life.

5. The Licensee should investigate the need and specific time duration for post-accident, long-term monitoring.

FRC concludes that, although not referenced by the Licensee, qualification documentation is available from the Foxboro Company and that the transmitter has a high likelihood of operating during a postulated LOCA, especially since the transmitter is enclosed and the amplifier is located outside containment. The NRC should judge the validity of the Licensee's contention that the reactor level transmitter's loss will not impair the safe shutdown of the plant.

LICENSEE RESPONSE:

> Reactor Vessel Water Level Transmitters (Item 43 of Enclosure 2)
>
> The reactor water level indicators, which are part of reactor protective system narrow range, initiate reactor shutdown (high and low water level), emergency core cooling system start (low water level), and containment isolation (low water level). This equipment performs these functions in a short time following a loss of coolant event. The application information for these transmitters is attached as Enclosure 6. The unique installation at LACBWR remotes the level transmitter amplifier out of the harsh environment.
>
> DPC will review the qualifications of this equipment against the specific vendor test data by April 1, 1981. If the type qualifications cannot be substantiated, replacement water level transmitters of current qualification where required for alternate core spray will be installed by June 30, 1982.

ENCLOSURE 6 [TO REFERENCE 23]

REACTOR VESSEL SAFETY SYSTEM WATER LEVEL TRANSMITTERS

MODEL

Water Level #1 T/613DM - MS2-0
Water Level #2 T/613DM - MS2-0
Water Level #3 E13DM Style B (No other Letters)

CASE

Water Level #1, Style E
Water Level #2, Style E
Water Level #3, Style B

CURRENT

Water Level #1, 2, 3 10-50 MA DC

AMP

Water Level #1, 2 Remote Amplifier N119LN
Water Level #3 Remote Amplifier N0141NL

BODY

Water Level #1, 2, 3, Body Material Stainless Steel
Cover-Cast Aluminum, Water Tight

CAPSULE

Water Level #1, 2, 3, Capsule 316SS A 62139 U102XF (Teflon)

ELECTRICAL CONNECTION

See attached diagram. Terminal Board V-113FZ Foxboro

SPECIAL MODEL

None

FRC EVALUATION:

The Licensee's response to the DITER stated that (i) this equipment performs safety functions in a short time interval after a LOCA; (ii) DPC will review evidence of qualification by April 1, 1981; and (iii) DPC will replace the transmitters required for alternate core spray by June 30, 1982 if type test data cannot support qualification. FRC notes that the Licensee has identified the specific model number of transmitter Nos. 1 and 2 as T/613DM-MS2-0; however, Reference 7 and Table 2 originally identified all three water level transmitters as model E13DM. FRC has therefore evaluated water level transmitters 1 and 2 (T/613DM-MS2-0) under Equipment Item 7A of this report.

With respect to the E13DM transmitter, the Licensee has not provided additional references as evidence of qualification. Therefore, the specific deficiencies identified in the DITER remain unchanged. The specific areas of deficiency cited were:

- o A manufacturer's installation instruction rather than a valid test report was submitted as evidence of qualification.
- o The effects of gasket and seal aging degradation have not been addressed by the Licensee for the transmitters' overall protective enclosure.
- o The exact relationship between the installed transmitter and the test specimen has not been established.
- o Transmitter aging degradation and qualified life have not been addressed by the Licensee.
- o The Licensee was to obtain the necessary qualification documentation from Foxboro Company.

After review of the Licensee's response, FRC notes the following:

- o The transmitter is totally enclosed in a watertight container and therefore submergence, spray, and pressure will have no effect on the units.
- o Test Report No. T3-1013 has established that the test chamber temperature/pressure profile under all steam conditions exceeded the postulated accident profile (with the exception of time duration as discussed in Appendix A) for 12 hours.
- o The accuracy of the E13DM transmitter as stated in Test Report No. T3-1013 under high temperature and steam conditions is acceptable.
- o The Licensee has provided details of the method of electrical connection (MI cable) for the transmitter enclosure. It appears that the possibility of steam entry into the enclosure is eliminated. Therefore, FRC finds this electrical connection at the enclosure acceptable.
- o The Licensee has stated that the LACBWR design and installation places amplifier (NO141NL) remote from the harsh environment. Because the amplifier is located in an area (control room) where the integrated radiation level is negligible, neither degradation of the top works (amplifier) due to radiation nor a proper test sequence is of concern.
- o The exact relationship between the installed transmitter and the appropriate test specimen has not been established. Various E13DM

models were tested (such as E13DM-ISAM2); however, the Licensee has stated E13DM- "no number." The test specimen transmitter had a special modification designation MCA/RRW which identified it for use under severe environments. The Licensee stated that the installed transmitters had no special modification. The test specimen E-capsule assemblies (Test Report No. T4-6040) and O-rings were different from those of the installed transmitter.

- The Licensee has not addressed aging degradation and qualified life.
- The Licensee has not addressed the need for post-accident, long-term monitoring.
- Enclosure 6 of Reference 23 indicates that a Foxboro V-113FZ terminal board is used to terminate the MI cable leads inside the transmitter enclosure. Evidence of qualification for this terminal board has not been provided.

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b because there is a high likelihood of operability due to the unique LACBWR installation. Short-term safety can therefore be achieved although complete evidence of qualification for this item is lacking. FRC concludes that the exact relationship between the installed transmitter and the appropriate test specimen has not been established, including special modifications for severe environments. In addition, the Licensee has not provided evidence of qualification for the Foxboro V-113FZ terminal board. Also, neither aging degradation nor qualified life has been addressed. The Licensee has stated that a qualification review will be conducted by April 1, 1981 and the transmitters required for alternate core spray will be replaced with fully qualified units by June 30, 1982 if the documentation cannot substantiate qualification.

The Licensee has stated that the transmitters associated with alternate core spray may be replaced. FRC recommends that all of the transmitters be replaced in order to provide adequate short-term and long-term safety functions.

4.5.2.2 Equipment Item No. 23
Temperature Detector T/C Located Inside Containment
Thermo Electric Model Ceramo J-116-G-304-00-20-1
Monitors Containment Building Temperature
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

The Licensee has stated that Type J iron constantan thermocouples have been provided for this containment application and that no exposed connections exist within containment. The manufacturer's literature states that the thermocouples have been tested to 50,000 psi external pressure and can operate to 1400°F. They have a 304 stainless-steel sheath, which should offer sufficient corrosion resistance. At a minimum, a qualification analysis should have been performed to demonstrate potential failure modes and to establish links with previous test results.

FRC has reviewed its files to determine if any previous testing has been conducted by FIRL and has been unable to find any test reports for Thermo Electric thermocouples. FRC was able to find a test report on Thermo Electric thermocouple wire, however, and noted that degradation was a problem. The Licensee should review the plant to determine if thermocouple extension wire is used in the plant for containment temperature monitoring and whether or not qualification documentation is available.

FRC concludes that the Licensee needs to provide thermocouple qualification documentation and to investigate the possible use of thermocouple extension wires.

LICENSEE RESPONSE:

Reactor Containment Building Thermocouples (Item 40 of Enclosure 2)

We have investigated the Type J iron constantan thermocouples used at LACBWR and have a continuous 304 stainless steel sheath for their entire length through the containment penetration (see Enclosure 4). Therefore, thermocouple extension wires are not used within the containment building.

We will have an analysis of the thermocouples used to verify qualification acceptability. The results of this analysis will be presented to the NRC by April 1, 1981.

FRC EVALUATION:

The Licensee has not furnished additional qualification documentation as evidence of the thermocouples' operability; instead, The Licensee has selected April 1, 1981 as a release date for this information.

The Licensee should provide evidence that the thermocouples will not degrade in a postulated LOCA environment, as well as a statement regarding qualified life.

The thermocouple extension wire concern has been alleviated because the Licensee states the wire is enclosed in a continuous 304 stainless steel sheath and no extension wires are used in containment.

FRC CONCLUSION:

These thermocouples are assigned to NRC Category IV.b because they have a reasonable likelihood of performing due to the simple design of the component but no qualification documentation has been provided.

4.5.2.3 Equipment Item No. 13
Terminal Blocks Located Inside Containment
Buchanan Model 218
(Licensee References 15 and 21)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

During the site visit by NRC and FRC representives on September 11 and 12, 1980, Licensee representatives stated that the watertight junction boxes located in containment enclosed the Buchanan terminal blocks. Subsequently, the Licensee transmitted a catalog cut of Model 218 terminal blocks and cited

Reference 21 for qualification. FRC has reviewed Reference 21 and finds that NQB type terminal blocks were tested and not the Model 218.

The Licensee should have the manufacturer confirm that testing demonstrates qualification in accordance with DOR Guidelines if testing has been conducted on the Model 218.

LICENSEE RESPONSE:

Terminal Blocks in Water Tight Junction Boxes

The terminal strips in five environmentally qualified junction boxes did not have qualification documentation available. DPC will replace these strips at the first outage of sufficient duration following the receipt of the replacement terminal strips. The replacements are Buchanan terminal strips, Models NQB-112 and NQB-106. The qualification test program is attached as Enclosure 3.

FRC EVALUATION:

The Licensee response [23] (Enclosure 3) contained, among other things, a specification for environmental tests for terminal blocks and fuse blocks but did not include a test report. However, FRC has reviewed FIRL Report F-C5143 on NQB type blocks and notes the following:

a. The terminal blocks were aged at 165°C for 950 hours, then irradiated to an exposure of 200 Mrd.

b. After thermal and radiation aging, the terminal blocks were subjected to vibration and seismic aging.

c. After seismic aging, the units were subjected to LOCA conditions in accordance with IEEE Std 323-74. Insulation resistance was satisfactory throughout the testing.

d. No submergence testing was performed.

The tests envelop the LaCrosse environmental conditions except for submergence. The tested enclosure did not have the same design as the LaCrosse enclosure, but testing of the LaCrosse enclosure was reviewed for the DITER and found satisfactory for the LaCrosse conditions.

FRC notes that the Licensee has not evaluated aging degradation or determined the qualified life of the installed equipment.

FRC CONCLUSION:

This equipment item is assigned to NRC Category IV.b. The Licensee has stated that the existing terminal blocks will be replaced by the Buchanan NQB type. When the modification has been completed and the Licensee has established the qualified life, this equipment can be assigned to NRC Category II.a.

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

4.6.1 Equipment Item No. 1
Electric Motors Located Inside Containment
Allis-Chalmers Model Type G, Class H, Silicoflex
High Pressure Core Spray Pumps 1A and 1B
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

The Licensee did provide some documents that discussed the type of motor as well as its operation in the high pressure core spray system (HPCS). As noted above, however, these submittals do not provide the necessary type of qualification documentation.

The Licensee stated that the HPCS pump motor is a 50-hp squirrel cage induction motor, which drives a positive displacement pump. It has Class H insulation, which normally has the ability to withstand radiation levels up to 100 Mrd although the SAR Figure 14.3 infinite dose rate is 1 Mrd. The motor is above the flood level so submergence is not a potential problem. The motor originally had a Class A-7 insulation, but it had been returned to its manufacturer, Allis-Chalmers, for modifications; it was then rewound to incorporate Class H insulation and given an epoxy Class H varnish treatment, which should increase the insulation's high temperature limit to 350°F. Also, while the motor was reworked at Allis-Chalmers, the grease was changed to lithium grease for higher temperature application. Temperature in the containment is expected to have a maximum peak temperature of 257°F; therefore, this temperature limit coupled with the motor's normal expected operating temperature rise would not likely exceed the 350°F limit. As mentioned previously, however, this motor lacks the positive testing program assurance that it could be reasonably expected to survive the postulated LOCA conditions and operate successfully. This motor is expected to operate for a maximum of 4 hours during the short-term period following the postulated LOCA and is additionally expected to operate during the long-term cooling period because the pump motors are sealed from the harsh environment.

In addition, the Licensee has stated that even if these motors would fail because of the severe containment environment, the plant cooling mode can rely upon a backup system, the low pressure core spray system (LPCS), to assist in mitigating a possible accident. A problem exists with this approach because the LPCS bypass valve (53-25-001) does not have a safety-related solenoid valve to allow the valve's opening for the long-term cooling mode. Therefore, the backup to the HPCS pump motor is not able to provide the necessary degree of assurance of its operability.

No data is currently available within the FRC files relating to Allis-Chalmer motors; therefore, FRC is unable to conclude that any type of qualification documentation exists that would demonstrate the motor's ability to operate under the harsh environment of a LOCA.

LICENSEE RESPONSE:

> These pumps are used for short-term emergency core cooling. They were modified for the harsh environment; however, testing qualification documentation is not available. A redundant system, the alternate core spray system, which has its pumps located outside the LOCA environment, is available.
>
> DPC will analyze the design of the high pressure core spray pump motors to assure their current applicability to the harsh environment. The results of this analysis will be furnished to the NRC by April 1, 1981.

FRC EVALUATION:

The Licensee has not furnished additional qualification documentation as evidence of the motors' operability but plans to supply the information by April 1, 1981.

The information provided by the Licensee to date has not identified the type of motor-lead splices, lead-to-cable splices, type of bearings, or the lubricant currently being used. The effects of the steam environment and radiation on these components were not reviewed to determine if age-related degradation would occur. A statement regarding the motor's expected qualified life was not provided.

FRC CONCLUSION:

These motors are assigned to NRC Category V because qualification documentation has not been provided. The Licensee should provide additional information relative to details of the motors' material construction and should analyze or test the effects of the environmental service conditions. Qualified life should be determined, and maintenance analysis should be presented to determine if abnormal wear could shorten the unit's qualified life. In regard to justification for interim operation, FRC has no technical objection to the Licensee's position as discussed in Appendix H, Section H-1.

4.6.2 Equipment Item Nos. 2, 3, and 4
Solenoid Valve Located Inside Containment
2: ASCO Model WPX8315B33
Actuates LPCS and MDS Vent Valves
3: ASCO Model HV202-301-4RG
Actuates HPSW Valve
4: ASCO Model HV202-924-4RG
Actuates Demineralized Water Valve
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

In Reference 7, the Licensee stated that the solenoids involved have Class H insulation rated for 356°F and that the housing is explosion-proof and uses Ml cable with sealed terminations. Reference 7 further states that the configuration enables the valve to withstand the LOCA environment with respect to temperature, pressure, humidity, and chemical attack, and that the infinite dose of less than 1 Mrd is below tolerances associated with Class H equipment. No test report was cited, and FRC is not aware of a test report on the valve model involved. FRC has reviewed Reference 7 and has the following comments:

a. The Guidelines require that complete and auditable records reflecting a comprehensive methodology be referenced and available for review for all Class 1E equipment. Type testing is the preferred method of qualification for Class 1E equipment located inside containment required to mitigate the consequences of a design basis event. A design specification is not sufficient. Qualification by type testing requires that the simulated environment in the test chamber envelop the specific service conditions identified. The type test is valid only if the test specimen and installed equipment have the same design, material, and production procedures. An analysis of the impact of deviations between the test specimen and installed equipment is an essential part of the qualification documentation.

b. From test reports that have been reviewed, FRC notes that failures of solenoid valves have occurred as a result of the lubricants, the springs, and the seat materials used. In addition, continuously energized Class H coils with explosion-proof housings have exhibited failures when subjected to LOCA conditions similar to those described for LaCrosse.

c. FRC understands that tests in accordance with IEEE Standards 323 and 382 have been conducted on NP series solenoid valves; however, test reports have not been reviewed.

In conclusion, the information provided by the Licensee is not adequate to establish whether the solenoid valves are satisfactory for the LaCrosse LOCA condition. It is suggested that the manufacturer be contacted for either specific tests on the installed models or analysis that would apply successful testing results for other valves to the installed models.

LICENSEE RESPONSE:

Item 2: Low Pressure Core Spray Valve

This valve functions to inject core spray directly into the high pressure core spray header without using the high pressure core spray pumps. When the differential pressure between the primary system and the reactor containment building is 30 psig or less and reactor water level of -12 inches, this valve will open and demineralized water from the overhead storage tank will flow by gravity at a rate of approximately 85 gpm to the high pressure core spray bundle.

While LACBWR feels that these Class H coils with explosion-proof housings served by mineral insulated cable are completely suitable for this environment, attempts to obtain qualification data will be more time-consuming and costly than replacement of the solenoid valve. DPC will therefore replace this item with a currently qualified model by June 30, 1982.

Interim operation with the existing equipment poses no risk as:

(1) The valve was originally selected for the harsh environment.

(2) The valve is in a fail-safe state (closed) - that is, it does not have to change position for the high pressure core spray to function; additionally there is an in-series check valve (53-26-001).

(3) Should the valve fail to open, a totally redundant core cooling system (alternate core spray and manual depressurization combination system) which is single failure proof is available.

(4) Use of the manual depressurization system would permit low pressure core spray through the valve if it failed in an open position.

Item 3: High Pressure Service Water Alternate Supply to the High Pressure Core Spray System

This valve functions to supply an additional water source to the high pressure core spray system. As the high pressure core spray system is intended as the short term emergency core cooling system and its water supply for this function is maintained in the overhead storage tank, this valve does not have a direct safety-related function.
Failure of this valve, which would lead to its changing state, would not cause a bypass of water from the overhead storage tank because of check valve (53-26-004) which is in series with the control valve.

Not withstanding the fact that use of this valve is not required, it does have a Class H coil with an explosion-proof housing and is served by mineral insulated cable and thus is judged suitable for the harsh environment.

LACBWR will, for consistency of certification on solenoid valves in this service, replace this valve with one having current qualification by June 30, 1982.

Item 4: Demineralized Water Supply to the Overhead Storage Tank

This valve functions to supply additional water to the overhead storage tank. As the high pressure core spray system is intended as the short-term emergency core cooling system and its water supply for this function is maintained in the overhead storage tank at all times, this valve does not have direct safety-related function. Failure of this valve, which would lead to its changing state, would not cause a diversion of water from the overhead storage tank because check valve No. 69-26-002, which is in series with the control valve, would prevent back flow.

Notwithstanding the fact that use of this valve is not required, it does have a Class H coil with an explosion-proof housing and is served by mineral insulated cable and thus is judged suitable for the harsh environment.

LACBWR will, for consistency of certification on solenoid valves in this service, replace this valve with one having current qualification by June 30, 1982.

FRC EVALUATION:

The Licensee is committed to replacing these solenoid valves for the services described.

The Licensee contends that either (i) the solenoid valve is not serving a safety-related function or (ii) if it is serving in a safety capacity, it will function in a fail-safe manner. Complete evidence of (ii) has not been provided. Neither solenoid valve failure analysis nor an operability test report has been presented. Similarly, neither maintenance inspection nor repair surveillance and analysis information has been made available.

FRC notes that the LPCS valve is highly important because the solenoid valve's function is to open and hold the valve between the overhead storage tank and the reactor, and in this way to provide an engineered safeguard system similar to the accumulator system on PWR plants. FRC also notes that the Licensee has not included in Reference 23 two of the solenoid valves that had been grouped with the LPCS valve as FRC Equipment Item No. 2 in the DITER· those actuating the MDS vent valves (G2-025-015 and -016). These solenoids had been listed in the first submittal [7].

FRC CONCLUSION:

This equipment is assigned to NRC Category V because of lack of evidence to demonstrate qualification. The Licensee is committed to replacing these solenoid valves by June 30, 1982. A general discussion of the justification for interim operation is presented in Section H.1 of Appendix H.

4.6.3 Equipment Item Nos. 5A and 6
Solenoid Valves Located Inside Containment
5A: ASCO Model 8300B9RF
6: ASCO Model 8300B9F
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

In Reference 7, the Licensee states that failure of these valves is not critical since a de-energization failure will result in positioning the process valve in the desired long-term position. The basis for this conclusion is the Licensee's analysis that chemical attack, temperature, and radiation would only affect the solenoid's coil, causing it to open or shut, either of which would de-energize the valve. Therefore, qualification of the valve would not be required.

FRC has reviewed the information provided by the Licensee in References 7 and 18 and has the following comments:

a. As noted in 3.3.2.1 above, FRC has reviewed tests on solenoid valves in which the LOCA conditions affected lubricants, springs, and seat materials as well as solenoid coils. FRC further notes that at least three solenoids of this model still had resilient seats in January 1979. Lubricants were not stated. It is suggested that the Licensee contact the manufacturer to establish whether the lubricant can be affected by the LOCA conditions so that it would cause sticking. (FRC notes that it is planned to change the Buna material.)

b. Subject to verification that the valve will not stick as a result of LOCA exposure, the Licensee should obtain NRC concurrence in the adequacy of the evaluations contained in Reference 7.

LICENSEE RESPONSE:

Item 5A: Shutdown Condenser Steam Inlet Valves

These solenoid valves control the parallel inlet valves to the shutdown condenser and manual depressurization system.

While LACBWR feels that these solenoids served by mineral insulated cable are completely suitable for this environment, attempts to obtain qualification data will be more time-consuming and costly than replacement of the solenoid valve. DPC will therefore replace this item with a currently qualified model by June 30, 1982.

Interim operation with the existing equipment poses no risk as:

(1) the valves were originally selected for the harsh environment, and

(2) the two parallel lines are totally redundant one to the other.

Item 5A: Isolation of Nonessential Demineralized Water System Loads in Containment

These solenoid valves control the valve which is closed upon a containment isolation signal to remove nonessential demineralized water loads within the Containment Building. One solenoid valve was previously identified in Reference (2); both valves are addressed in this letter. As the water supply for the overhead storage tank is maintained at all times in an amount adequate for the short-term emergency core cooling system purpose for which the high pressure core spray system is intended, the valve is not directly safety-related.

Notwithstanding the fact that use of this valve is not required, Dairyland Power Cooperative feels it is suitable for the harsh environment.

LACBWR will, for consistency of certification on solenoid valves in this service, replace this valve with one having current qualification by June 30, 1982.

Item 6: Isolation of Nonessential High Pressure Service Water System Loads in Containment

These solenoid valves control the valve which is closed upon a containment isolation signal to remove nonessential high pressure service water system loads within the containment building.

As the water supply for the high pressure core spray system to perform its design function as a short-term emergency core cooling system is maintained at all times in the overhead storage tank, this valve is not directly safety-related.

Notwithstanding the fact that use of this valve is not required, Dairyland Power Cooperative feels it is suitable for the harsh environment.

LACBWR will, for consistency of certification on solenoid valves in this service, replace this valve with one having current qualification by June 30, 1982.

FRC EVALUATION:

The Licensee contends that either (i) the solenoid valve is not serving a safety-related function, or (ii) the solenoid valve is backed up by other systems which can perform the shutdown function.

The solenoid valves have three applications:

1. shutdown steam inlet valves for shutdown condenser and manual depressurization system
2. isolation of nonessential demineralized water systems loads
3. isolation of nonessential high pressure service water system loads in containment.

The first case is the most critical, because it allows the manual depressurization system to operate. This solenoid must operate for several hours after the initiation of a postulated accident, and evidence of its function and operability has not been provided.

FRC notes that the Licensee is committed to replacing these solenoid valves for the services described by the Licensee.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because of lack of evidence to demonstrate qualification. The Licensee is committed to replacing these solenoids by June 30, 1982.

4.6.4 Equipment Item No. 7A
Level Transmitter Located in Containment
Foxboro Model T/613DM-MS2-0
Reactor Water Level Transmitters:
Water Level No. 1, Narrow Range, 50-42-302
Water Level No. 2, Narrow Range, 50-42-303
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

(Discussed in DITER as Equipment Item 7, Section 3.3.3.2; see 4.5.2.1 of this report.)

LICENSEE RESPONSE:

The reactor water level indicators, which are part of the reactor protective system narrow range, initiate reactor shutdown (high and low water level), emergency core cooling system start (low water level), and containment isolation (low water level). This equipment performs these functions in a short time following a loss of coolant event. The application information for these transmitters (as required by Page 17 and 21 of Reference 2) is attached as Enclosure 6. The unique installation at LACBWR remotes the level transmitter amplifier out of the harsh environment.

DPC will review the qualifications of this equipment against the specific vendor test data by April 1, 1981. If the type qualifications cannot be substantiated, replacement water level transmitters of current qualification where required for alternate core spray will be installed by June 30, 1982.

ENCLOSURE 6 [TO REFERENCE 23]

REACTOR VESSEL SAFETY SYSTEM WATER LEVEL TRANSMITTERS

MODEL

Water Level #1 T/613DM - MS2-0
Water Level #2 T/613DM - MS2-0
Water Level #3 E13DM Style B (No other Letters)

CASE

Water Level #1, Style E
Water Level #2, Style E
Water Level #3, Style B

CURRENT

Water Level #1, 2, 3 10-50 MA DC

AMP

Water Level #1, 2 Remote Amplifier N119LN
Water Level #3 Remote Amplifier N0141NL

BODY

Water Level #1, 2, 3, Body Material-Stainless Steel
Cover-Cast Aluminum, Watertight

CAPSULE

Water Level #1, 2, 3, Capsule 316SS A62139 U102XF (Teflon)

ELECTRICAL CONNECTION

See attached diagram. (Terminal Board V-113FZ Foxboro)

SPECIAL MODEL

None

FRC EVALUATION:

The Licensee's response stated that (i) this equipment performs safety functions in a short time interval after a LOCA; (ii) DPC will review evidence of qualification by April 1, 1981; and (iii) DPC will replace the transmitters required for alternate core spray by June 30, 1982 if type test data cannot support qualification. FRC notes that the Licensee has identified the specific model number of transmitter Nos. 1 and 2 as T/613DM-MS2-0; however, Reference 7, Table 2 originally identified all three water level transmitters as model E13DM (FRC originally included these transmitters in the discussion under Equipment Item 7, Section 3.3.3.2 of the DITER).

The Licensee did not provide qualification documentation for this equipment (Model T/613DM-MS2-0) as required by the DOR Guidelines. The Guidelines require that complete and auditable records reflecting a comprehensive qualification methodology be referenced for review for all Class 1E equipment.

FRC has reviewed its files to determine if any previous testing has been conducted on this equipment item and has been unable to find any test reports or other documentation that would substantiate qualification.

FRC concludes that these transmitters are not qualified because no evidence of qualification has been made available.

FRC CONCLUSION:

This equipment item is assigned to NRC Category V because qualification documentation has not been made available. The Licensee has stated that a qualification review will be conducted by April 1, 1981 and the transmitters required for alternate core spray will be replaced with fully qualified units by June 30, 1982 if the documentation cannot substantiate qualification.

In addition, the Licensee has stated that the transmitters associated with alternate core spray may be replaced. FRC recommends that all the transmitters should be replaced in order to provide adequate short-term and long-term safety functions.

4.6.5 Equipment Item No. 8
Switchgear Located Inside Containment
Allis-Chalmers Valve Line MCC
Switchgear for Demineralized Water Pump 1B
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

Because the switchgear is located inside containment (even though the pump motor to which it is designated is located in the turbine building) and is susceptible to flooding, the Licensee has stated in Reference 7 that its loss is basically inconsequential since a backup demineralized water pump (1A) will be available. In addition, the Licensee stated that a temporary power connection could be provided to demineralized water pump 1B if required; however, details of how this electrical connection would be performed safely were not provided.

FRC has reviewed its qualification documentation files and has found no test reports for Allis-Chalmer switchgear. FRC therefore concludes that the switchgear would fail under LOCA conditions and that the NRC must review the contention of the Licensee that its failure is inconsequential to continued operation of the demineralized water system.

FRC suggests that the switchgear be relocated outside containment in a mild environment.

LICENSEE RESPONSE:

Demineralized Water Transfer Pump Switchgear

This switchgear provides power to the 1B demineralized water transfer pump. This equipment is not safety-related, as the short-term emergency core cooling system (the high pressure core spray system) already has its water supply for this function in the overhead storage tank. The long-term emergency core cooling system (the alternate core spray system) does not use demineralized water. Operationally, the 1B demineralized water transfer pump has a redundant pump which has switchgear located outside the containment building. Additionally, the 1B pump itself is located outside the containment building. No action by DPC is required on this switchgear, and it should be removed from the list.

FRC EVALUATION:

The Licensee has stated that the equipment for the demineralized water transfer system is not safety-related. FRC would agree with this statement if the high pressure core spray system, the manual depressurization system, and the overhead storage tank system were shown to be qualified and safety-related systems, as stated in Appendix H, Section H-1. At this time, however, evidence of qualification of these safety systems has not been provided and the importance of the demineralized water transfer pumps as a viable backup system should not be underestimated in light of their versatility in supplying cooling water to the surface condenser and water to the overhead storage tank.

The Licensee should demonstrate that the loss of the switchgear would not degrade any safety-related power distribution systems.

FRC continues to recommend the relocation of the switchgear to a nonharsh environment.

FRC CONCLUSION:

This switchgear is assigned to NRC Category V because evidence has not been found to show that the switchgear could survive containment environmental service conditions. The Licensee believes the equipment should not be considered safety-related and should be exempt from qualification. If evidence is produced to relieve the concerns (documented in Appendix H, Section H-1) relating to the qualifications of the ACS and MDS systems, then the equipment can be reassigned to NRC Category III.

4.6.6 Equipment Item No. 10
Motor Starter Located in Turbine Building
Cutler Hammer Model K646676A
Controls Power to MOV 38-30-002 Controlling ACS Valve
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

Reference 11 calculated the predicted rise in the turbine building pressure from a postulated MSLB and arrived at a 3 psi pressure increase. A subsequent calculation showed the pressure increase to be only 0.17 psi, as indicated in Appendix G, Item 7. The relative humidity is assumed to climb to 100% and the temperature to 213°F. The Licensee contends that the effects of this excursion are negligible since the building volume is great and the steam release should only occur for a few seconds. The dc motor starter is in close proximity to a 10-inch main steam line. It is housed in a gasketed enclosure, and the Licensee has stated in Reference 7 that even if the alternate core spray (ACS) valve were made inoperable, the HPCS system would be available to back it up.

FRC has reviewed its qualification documentation file and has found no test reports for Cutler Hammer dc motor starters.

FRC concludes that there is presently no evidence to show that the equipment meets the Guidelines' requirements for components in harsh environments and that the Licensee should provide either type test or analysis information, which could demonstrate operability of the motor starter for the short-duration service conditions.

LICENSEE RESPONSE:

> This motor starter controls one of the redundant alternate core spray valves located within the turbine building. As the event which could create a hostile environment in this area does not require use of the alternate core spray system [5], there is no requirement for this equipment to be qualified.

FRC EVALUATION:

LACBWR Safeguards Report, pages 14-30, indicates, with respect to the steam line system outside containment, that "it is doubtful that any fuel melting would occur since feedwater flow to the primary system would continue after a rupture and would tend to restore the water level, which would prevent core damage." Such reliance on the feedwater system for decay heat removal is not supported by documentation indicating that the feedwater system is designed to be safety-related. Similarly, other systems which could remove decay heat (e.g., the shutdown condenser or decay heat removal system) are neither safety systems nor provided with a safety-related source of cooling water (component cooling, HP service water, demineralized water).

Reliance for accident mitigation cannot be placed on the operation of systems which have not been designed and constructed to engineered safety feature standards. In the present case, the validity of this principle is evident upon consideration of a postulated condition involving the mitigation of a primary system rupture outside containment. Under such circumstances, a seismically unqualified system should not be relied on to mitigate the consequences of such a rupture. The ACS system is the only one identified by the Licensee as an engineered safety feature system to perform long-term core decay heat removal following an accident.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because no documentation for qualification has been provided by the Licensee. This equipment should be qualified for the environmental conditions to which it is exposed. The Licensee should provide a statement on the qualified life of the equipment in accordance with Section 4.1.3.

4.6.7 Equipment Item No. 18
Solenoid Valve Located Inside Containment (MSIV)
ASCO Model X-8344
Actuates Main Steam Isolation Valve
(Licensee Reference TR-6)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

In the table for the hydraulic valve accumulator system of Reference TR-6, the Licensee identifies solenoid valves for MSIV application and states that a LOCA causes loss of power, which, in turn, causes the valves to fail in the position that closes the MSIV.

Since there is no model or manufacturer listed for the solenoid valve, FRC cannot independently confirm that the valve will fail as indicated. The Licensee should provide test data or analysis that the valve will fail as identified so that NRC can evaluate the safety issue discussed in Reference TR-6.

LICENSEE RESPONSE:

Reactor Building Main Steam Isolation Valve Control Solenoids

This dual solenoid control valve controls the containment isolation valve in the main steam line. The control assembly which includes the solenoid valves was scheduled previously by LACBWR for replacement. Solenoid valves with current qualification will be utilized. This replacement will be installed by June 30, 1982.

Operation of this system in the interim with the currently installed equipment poses no significant risk as the MSIV closure is accomplished immediately upon a low reactor water level which would indicate a loss of coolant accident leading to a harsh environment.

There is no requirement to reopen the MSIV in a post LOCA condition. Additionally, a redundant valve exists in the main steam line located outside the containment building which can be closed as a backup. The valve is electric motor operated, controlled remotely from the control room with the provision for local manual closing. It is powered from 480-volt Essential Bus 1A.

FRC EVALUATION:

The Licensee has indicated that these valves have been scheduled for replacement and has now committed to replace them with qualified units by June 30, 1982.

FRC CONCLUSION:

This solenoid valve is assigned to NRC Category V because no qualification documentation has been provided. FRC has no technical objections to the Licensee's systems evaluation for interim operation. The Licensee should establish a conservative qualified life for the replacement units (see Section 4.1.3).

4.6.8 Equipment Item Nos. 20A, 20B, and 21
Limit Switches and Solenoid Valves Located Inside Containment
20A: ASCO Model LM831612
20B: Barksdale Model 178350AC2A1
21: Limit Switches, Manufacturer and Model Not Stated
Actuates Containment Ventilation Intake and Exhaust Isolation Valves and Provides Position Indication
(Licensee Reference TR-6)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

In the table for the reactor building heating ventilation and air conditioning system of Reference TR-6, the Licensee identifies limit switches and solenoid valves for ventilation inlet damper application and states that a LOCA causes loss of power, which, in turn, causes the valves to fail in the position that closes the dampers.

Since there is no model or manufacturer listed for the solenoid valve, FRC cannot independently confirm that the valve will fail as indicated. The Licensee should provide test data or analysis that the valve will fail as identified so that the NRC can evaluate the safety issue discussed in Reference TR-6.

LICENSEE RESPONSE:

Reactor Containment Building Ventilation

These control valves are closed to isolate the containment building from the outside environment. Closure is effected by removing the air through closure of the solenoid valves (2 per control valve).

DPC has previously provided (Reference 3) a commitment to replace the solenoid valves with currently qualified equipment. It (Reference 3, NRC Question 5) has also described the functioning of the solenoid valves to ensure environmental adequacy. The containment isolation is accomplished by two valves in series on the inlet and the exhaust; the limit switches provide no control function and are for indication only.

Interim operation with the existing solenoid valves poses no risk as:

(1) The existing solenoid valves were originally selected for this application with Class F and Class H coils.

(2) Functioning of these valves occurs immediately upon a LOCA and no further use is required in a post-accident situation.

(3) Replacement valves have been ordered and will be installed prior to June 30, 1982.

FRC EVALUATION:

The Licensee is committed to replacing these solenoids, and presumably their associated limit switches, with qualified equipment by June 30, 1982.

Replacement of the limit switches has not been addressed. FRC believes that the limit switches provide indication that the valves has closed upon receipt of a containment isolation signal. This information is important in the mitigation of potential accidents because the operators must know whether or not containment isolation valves have performed their isolation function. Also, some of the valves may have to be reopened during the later phases of accident mitigation.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because no qualification documentation has been provided. The Licensee is committed to replacement of these solenoids by June 30, 1982. FRC has no technical objections to the Licensee's evaluation of interim operation. The Licensee should establish a conservative qualified life for the replacement units (see Section 4.1.3).

4.6.9 Equipment Item Nos. 22A and 22B
Limit Switches and Solenoid Valves Located Inside Containment
22A: ASCO Model 8300B9RF
22B: Limit Switches, Manufacturer and Model Not Stated
(Licensee Reference TR-6)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

In the table for the reactor building heating ventilation and air conditioning system of Reference TR-6, the Licensee identifies limit switches and solenoid valves for exhaust damper application and states that a LOCA causes loss of power, which, in turn, causes the valves to fail in the position that closes the damper.

Since there is no model or manufacturer listed for the solenoid valve, FRC cannot independently confirm that the valve will fail as indicated. The Licensee should provide test data or analysis that the valve will fail as identified so that NRC can evaluate the safety issue discussed in Reference TR-6.

LICENSEE RESPONSE:

> This valve serves as a containment isolation valve in the 4-inch containment vessel offgas vent header. The control valve itself is a mechanical component, discussed in Reference (3), and should be removed from this qualification program. The solenoid valves (Items 26 and 27) are designed to operate in this environment. Attempts to obtain classification data will be more time-consuming and costly than replacement. However, DPC will replace these items with currently qualified valves by June 30, 1982.
>
> Interim operation with the existing solenoid valves poses no risk as:
>
> (1) The valves were originally selected for the harsh environment.
>
> (2) The valves will be required to operate only once immediately upon a LOCA; once the control valve is closed, no reopening in a post accident situation is required.
>
> (3) The control valve has redundant solenoid valves.
>
> (4) The control valve has a redundant isolation valve outside of the containment building which performs the same function.

FRC EVALUATION:

The Licensee is committed to replacing these solenoid valves with the qualified equipment by June 30, 1982.

Replacement of the limit switches has not been addressed. FRC believes that the limit switches provide indication that the valves have closed upon receipt of a containment isolation signal. This information is important in the mitigation of potential accidents because the operators must know whether or not containment isolation valves have performed their isolation function. Also, some of the valves may have to be reopened during the later phase of accident mitigation.

FRC CONCLUSION:

This equipment is assigned to NRC Category V because qualification documentation has not been provided. The Licensee is committed to replacement of these solenoid valves by June 30, 1982. FRC has no technical objections to the Licensee's evaluation of interim operation. The Licensee should establish a conservative qualified life for the replacement units (see Section 4.1.3).

4.7 NRC Category VI
EQUIPMENT FOR WHICH QUALIFICATION IS DEFERRED

The equipment items in this section have been addressed by the Licensee in the equipment environmental qualification submittal; 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.

4.7.1 Equipment Item No. 24
Radiation Monitor Located Inside Containment
General Atomic, Electronic Systems Division, Model RD-23
Monitors Containment Gamma Radiation
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

The Licensee provided the manufacturer's product literature to illustrate the monitor's basic capabilities According to General Atomic, the unit is designed for 350°F, 70 psig, and 100% relative humidity, but there is no indication of chemical spray resistance capability. From a design standpoint, the unit probably could meet the accident condition profile; however, the duration of the profile curve has yet to be determined.

FRC has reviewed its files for previous qualification test information and has not found any available.

The Licensee needs to provide qualification information relative to type testing of the radiation monitor or analysis of previous model testing programs that General Atomic may have conducted.

LICENSEE RESPONSE:

Post Accident High Range Radiation Monitors

Installation of these monitors has commenced. They have been procured to comply with a Three Mile Island Short Term Lessons Learned requirement. Installation of these monitors is required by October 1, 1981. Complete qualification data will be supplied to the NRC at that time.

FRC EVALUATION:

The Licensee states that qualification documentation for the high range radiation monitors will be made available by October 1, 1981 when the units are installed.

FRC CONCLUSION:

This equipment is assigned to NRC Category VI because no documentation for qualification has been submitted and the EEQ review of the equipment can be deferred until after February 1, 1981 under the terms of Section 2.2.5 (TMI Action Plan).

4.7.2 Equipment Item No. 25
Radiation Monitor and Sampler Located Inside Containment
Manufacturer Not Stated
Samples Containment Air for Radiation
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

LICENSEE RESPONSE:

Reactor Containment Building Air Monitor System

The monitor system functions to indicate gaseous and particulate levels in the containment ventilation system during operation. It provides a high radiation closure signal to the containment ventilation system. The closure actuation setpoints have been established at not more than 5 times background, thus activation of isolation can occur following any primary system leak as detected by the immediate particulate monitor within approximately 1.5 minutes.

Small leaks would be first indicated on the primary system leak detection system by humidity and radiation conditions, which is expected to cause operator action to isolate containment ventilation prior to reaching the closure activation setpoint for isolation by the subject monitor.

The LOCA environment from this leak would not be severe enough in that time span to affect the monitor. If, however, the monitor were affected, its failure mode is to isolate containment. Once isolated, the post accident procedures do not require the reopening of containment ventilation. Due to this design and the redundancy of closure signals from high primary system pressure, high containment building pressure and low reactor water level, no environmental qualification is required for this monitor.

FRC EVALUATION:

The Licensee contends that the containment radiation monitor and sampler located inside containment should be exempt from qualification because its design is fail-safe and redundant containment isolation signals are available inside the plant. However, the monitors require qualification in accordance with NUREG-0578 and -0737. As outlined in Section 2.2.5 of this TER, the NRC requires a pre-implementation review which the Licensee should submit by February 1, 1981.

FRC CONCLUSION:

The radiation monitor and sampler are assigned to NRC Category VI because qualification documentation has not been submitted and the EEQ review of the equipment can be deferred until after February 1, 1981 under the terms outlined in Section 2.2.5 (TMI Action Plan).

4.7.3 Equipment Item No. 5B
Solenoid Valve Located in Pipe Tunnel
ASCO Model 8300B9RF
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

In Reference 7, the Licensee states that failure of these valves is not critical since a de-energization failure will result in positioning the process valve in the desired long-term position. The basis for this conclusion is the Licensee's analysis that chemical attack, temperature, and radiation would only affect the solenoid's coil, causing it to open or shut, either of which would de-energize the valve. Therefore, qualification of the valve would not be required.

FRC has reviewed the information provided by the Licensee in Reference 7 and 18 and has the following comments:

a. As noted in 3.3.2.1 above, FRC has reviewed tests on solenoid valves in which the LOCA conditions affected lubricants, springs, and seat materials as well as solenoid coils. FRC further notes that at least three solenoids of this model still had resilient seats in January 1979. Lubricants were not stated. It is suggested that the Licensee contact the manufacturer to establish whether the lubricant can be affected by the LOCA conditions so that it would cause sticking. (FRC notes that it is planned to change the Buna material.)

b. Subject to verification that the valve will not stick as a result of LOCA exposure, the Licensee should obtain NRC occurrence in the adequacy of the evaluation contained in Reference 7.

LICENSEE RESPONSE:

This valve serves as a containment isolation valve on the four-inch containment vessel offgas vent header. The valve is not located inside the containment building, but in a portion of the tunnel. The valve is subjected to a mild environment; therefore, no qualification for a harsh environment is required. The isolation function of the control valve is totally redundant to the inside containment isolation valve. No action by DPC is required on this solenoid valve, and it should be removed from the list.

FRC EVALUATION:

The Licensee has stated that the solenoid valve is located in a mild area for the accident it is intended to mitigate. The review of this mild area equipment can, therefore, be deferred in accordance with Section 2.2.3.

FRC CONCLUSION:

This equipment is assigned to NRC Category VI because it is located in a mild area for the accident it is intended to mitigate. Its review is deferred until after February 1, 1981 in accordance with Section 2.2.3.

4.7.4 Equipment Item Nos. 14A and 14B
Pressure Transmitter Located in Pipe Tunnel
Foxboro Co.
14A: T611GM
14B: T631-2AS
Containment Building Pressure
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

These transmitters are located outside containment in the steam tunnel where they are exposed to the short-duration MSLB conditions. They are needed for the accident and post-accident monitoring system and have been enclosed in a weatherproof housing and tested to 52 psig pressure.

FRC has reviewed its files to determine if any previous testing has been conducted by Franklin Institute Research Laboratories and has not found any test reports.

FRC concludes that the Licensee will have to furnish the documentation for these transmitters.

LICENSEE RESPONSE:

Containment Building Post Accident Pressure and Water Level Indicator Transmitters

Two containment water level indicators and two containment pressure indicators are required for post accident information when a LOCA has occurred and containment personnel entry is not possible. This equipment is not required for steam line break outside containment; however, their watertight housings are judged to withstand the resulting short-term hostile environment. Therefore, documentation on this equipment is not required.

FRC EVALUATION:

The Licensee has submitted evidence that the area where the safety-related transmitters are located is mild for the accident condition that the device is designed to mitigate. The review of the transmitter's substantiating qualification is therefore deferred until after February 1, 1981 as discussed in Section 2.2.3.

FRC does not agree that the transmitter can be considered exempt, because the device must continue to perform its safety function of correctly transmitting a containment pressure signal in the event of a MSLB outside containment. The Licensee should provide additional information to demonstrate that the possible failure of the transmitter will not result in the degradation of the associated safety-related electrical circut when the transmitter is exposed to harsh conditions of a MSLB/HELB outside the containment.

A statement concerning qualified life needs to be provided by the Licensee. Also, maintenance records for the transmitter should be reviewed and summarized to determine if any abnormal difficulties have been experienced which could limit the qualified life.

There is a concern that failure of the containment pressure transmitter during a MSLB or HELB outside of the containment could result in the inadvertent actuation of the containment spray system by the plant operator because of an incorrect response by an unqualified instrument. This premature deployment of the containment spray system could result in vacuum conditions inside containment.

FRC CONCLUSION:

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

4.7.5 Equipment Item No. 15
Level Transmitter Located in Pipe Tunnel
Foxboro Co. Model T613-DM
Monitors Containment Building Water Level
(Licensee reference not cited)

ORIGINAL TEXT TAKEN FROM DRAFT INTERIM TECHNICAL EVALUATION REPORT:

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

This transmitter is designated for use as an accident/post-accident monitoring system device and has been provided with a weatherproof NEMA 3 housing.

FRC has reviewed its files to determine if any previous testing has been conducted by FIRL and has been unable to find any test reports for qualification.

FRC concludes that the Licensee will have to furnish the qualification documentation for these transmitters.

LICENSEE RESPONSE:

Containment Building Post Accident Pressure and Water Level Indicator Transmitters

Two containment water level indicators and two containment pressure indicators are required for post accident information when a LOCA has occurred and containment personnel entry is not possible. This equipment is not required for steam line break outside containment; however, their watertight housings are judged to withstand the resulting short-term hostile environment. Therefore, documentation on this equipment is not required.

FRC EVALUATION:

The Licensee has submitted evidence that the area where the safety-related level transmitters are located is relatively mild for the accident condition that the device is designed to mitigate. The review of the transmitter's substantiating qualification is therefore deferred until after February 1, 1981 as discussed in Section 2.2.3 for equipment located in mild area.

The Licensee should provide additional information to demonstrate that the possible failure of the level transmitter will not result in the degradation of the associated safety-related electrical circuit when the device is exposed to the harsh conditions of a MSLB/HELB outside the containment.

A statement concerning qualified life needs to be provided by the Licensee. Also, maintenance records for the transmitter should be reviewed and summarized to determine if any abnormal difficulties have been experienced which could limit the qualified life.

The Licensee has not provided qualification documentation for this equipment.

FRC CONCLUSION:

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

4.8 SUMMARY OF THE EVALUATION

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

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

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

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

L = A limiting factor with respect to qualification in that the qualified life has 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 Satisfies All Applicable Requirements for the Life of the Plant	1
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	0
II.b	Equipment Satisfies All Applicable Requirements With the Exception of Qualified Life Provided That Specific Modifications Are Made	0
II.c	Equipment Does Not Meet All Applicable Requirements; However, Deviations Are Judged Acceptable With the Exception of Qualified Life	4
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	3
V	Equipment is Unqualified	15
VI	Equipment Qualification is Deferred	6
		31

Table 4-2

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103

EQUIPMENT ENVIRONMENTAL QUALIFICATION

SEP PLANTS

FRC TASK	REACTOR TYPE	PLANT NAME	PAGE
02. 203	BWR	LaCrosse	1

PROJECT	UTILITY
02G-C5257-01	Dairyland Power Coop.

DOCKET	NRC TAC	DATE/ENGINEER
50-409	12382	4-15-81 Ja

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER																		
	1	2	3	4	5A	5B	6	7A	7B	8	9	10	11	12	13	14A	14B	15	16
GUIDELINE REQUIREMENTS.	(DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)																		
EVIDENCE OF QUALIFICATION	X	X	X	X	X	X	X	X	X	X	L	X	L		X	X	X	X	L
RELATIONSHIP TO TEST SPECIMEN								X											
AGING DEGRADATION EVALUATED	X	X	X	X	X		X	X			L				X				L
QUALIFIED LIFE ESTABLISHED	X	X	X	X	X		X	X			L			L	X				L
PROGRAM TO IDENTIFY AGING	X	X	X	X	X		X	X			L			L	X				L
QUAL. FOR STEAM EXPOSURE																			
PEAK TEMPERATURE ADEQUATE																			
PEAK PRESSURE ADEQUATE																			
TEST DURATION ADEQUATE																			
REQUIRED PROFILE ENVELOPED																			
QUAL. FOR SUBMERGENCE															X				
QUAL. FOR CHEMICAL SPRAY																			
QUAL. FOR RADIATION																			
BETA RADIATION CONSIDERED																			
TEST SEQUENCE																			
TEST DURATION (1 HOUR + FUNCTION)																			
QUANTITY OF EQUIPMENT																			
EQUIPMENT INSPECTED AT SITE																			
QUALIFICATION CATEGORY,	(O — CATEGORY DESIGNATION)																		
I-A. QUAL. FOR PLANT LIFE																			
I-B. QUAL. BY JUDGEMENT																			
II-A. QUAL. FOR < PLANT LIFE																			
II-B. QUAL. PENDING MODIFICATION																			
II-C. QUAL. < PLANT LIFE/FRC REVIEW													O	O					O
III. EXEMPT FROM QUAL.																			
IV-A. QUAL. TEST SCHEDULE																			
IV-B. QUAL. NOT ESTABLISHED									O						O				
V. EQUIP. NOT QUALIFIED	O	O	O	O	O		O	O		O		O							
VI. QUAL. IS DEFERRED						O										O	O	O	
REPLACEMENT SCHEDULE		R	R	R	R		R								R				
	M	SV	SV	SV	SV	SV	SV	LT	LT	SG	MVA	MS	C	XS	TB	PT	PT	LT	MVA

Table 4-2 (Cont.)

Franklin Research Center A Division of The Franklin Institute The Benjamin Franklin Parkway, Phila., Pa. 19103	FRC TASK 02.203	REACTOR TYPE BWR	PLANT NAME LaCrosse	PAGE 2
EQUIPMENT ENVIRONMENTAL QUALIFICATION	PROJECT 02G-C5257-01		UTILITY Dairyland Power Coop.	
SEP PLANTS	DOCKET 50-409	NRC TAC 12382	DATE/ENGINEER 4-15-81 ga	

SUMMARY REVIEW	EQUIPMENT ITEM NUMBER											
	17	18	19A	19B	20A	20B	21	22A	22B	23	24	25
GUIDELINE REQUIREMENTS.	(DESIGNATIONS: X — DEFICIENCY, L — LIMITING CONDITION)											
EVIDENCE OF QUALIFICATION		X			X	X	X	X	X	X		
RELATIONSHIP TO TEST SPECIMEN												
AGING DEGRADATION EVALUATED		X			X	X	X	X	X			
QUALIFIED LIFE ESTABLISHED		X			X	X	X	X	X			
PROGRAM TO IDENTIFY AGING		X			X	X	X	X	X			
QUAL. FOR STEAM EXPOSURE												
PEAK TEMPERATURE ADEQUATE												
PEAK PRESSURE ADEQUATE												
TEST DURATION ADEQUATE												
REQUIRED PROFILE ENVELOPED												
QUAL. FOR SUBMERGENCE												
QUAL. FOR CHEMICAL SPRAY												
QUAL. FOR RADIATION												
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	O											
I-B. QUAL. BY JUDGEMENT												
II-A. QUAL. FOR < PLANT LIFE												
II-B. QUAL. PENDING MODIFICATION												
II-C. QUAL. < PLANT LIFE/FRC REVIEW												
III. EXEMPT FROM QUAL.			O	O								
IV-A. QUAL. TEST SCHEDULE												
IV-B. QUAL. NOT ESTABLISHED										O		
V. EQUIP. NOT QUALIFIED		O			O	O	O	O	O			
VI. QUAL. IS DEFERRED											O	O
REPLACEMENT SCHEDULE		R			R	R	R	R	R			
	P	SV	SV	SV	SV	SV	SV	SV	SV	TE	RE	RE

5. CONCLUSIONS

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

The DOR Guidelines require the Licensee to have ongoing programs to review surveillance and maintenance records in order to assure that safety-related equipment that exhibits age-related degradation 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 a more detailed analysis is presented in Appendices F, G, and H.

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

Section 4.1 of this report, Methodology Used by the Licensee, reviews several concerns not adequately treated by the Licensee. The concerns range from disagreements over what constitutes safety systems to which equipment actually requires qualification documentation. A number of the Licensee's action items in the DITER's Appendix F were not completely resolved. In addition, the Licensee did not submit detailed supporting calculations to demonstrate that the results obtained from its long-term heat sink thermodynamic equation were accurate. Additional calculations are needed (Appendix G). Qualification documentation was not available for several equipment items, and the Licensee has committed to providing this information by April 1, 1981. However, this was 5 months after the deadline of November 1, 1980 when all documentation was to have been submitted.

Several of the plant's safety-related equipment items were not initially so identified by the Licensee. FRC discussed these items in the Conclusion section of the DITER to which the Licensee responded. FRC comments on the Licensee's response for most of these items are presented in Appendix I.

For some equipment items, as noted in this report, the Licensee should provide additional justification, such as maintenance records, analysis, or other test result information, in order to substantiate operability of these equipment items during and after a postulated accident condition.

The present review of the safety-related electrical equipment for the LaCrosse Boiling Water Reactor included only the investigation of equipment located in the "harsh environment" areas (i.e., containment and turbine building) and needed to ensure hot shutdown of the plant. The EEQ review of equipment items located in "mild" areas, and of those which are needed to bring the plant to a cold shutdown condition, has been deferred by the Licensee until after February 1, 1981.

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APPENDIX A - ENVIRONMENTAL SERVICE CONDITIONS

Since the issuance of the Draft Interim Technical Evaluation Report, the Licensee has provided additional information [23] regarding environmental service conditions both inside and outside containment. The post-accident containment temperature profile curve was revised to take into account the approximate effects of employing a new cooling system to provide a long-term heat sink for the containment. This future long-term cooling system would use the existing component cooling water piping after it was upgraded to a safety class system. The Licensee did not provide additional long-term heat sink detailed calculations for the plant's existing long-term cooling method (the technique of containment dome insulation removal followed by spraying the exterior of the containment with cooling water). Such analyses are needed to demonstrate the ability of the existing method to cool down both this containment and the reactor to a cold shutdown state (see Section G-2 of Appendix G).

The Licensee provided recent results of a turbine building MSLB and main feedwater line break analysis, which are shown in Figures A-11 through A-13.

The specific environmental zones within the plant are shown on LaCrosse Boiling Water Reactor plant General Arrangement drawings (Figures A-2 through A-6).

The environmental service conditions for temperature and pressure inside containment are shown on Figure A-1. It should be noted that the high temperature and pressure values on the higher profile envelope curve do not decrease significantly with time after a postulated accident. The Licensee was reviewing this conditon with the expectation of providing an alternate safety-related containment cooling mechanism. The final Licensee information submittal [23] presented general data (depicted on Figure A-1) for the containment temperature profile under conditions that would exist if the component cooling water system was employed to achieve long-term cooldown.

The Licensee has noted that the environment within the control room and electrical equipment room is not affected by a LOCA [7]. However, in Reference 23, the Licensee provided data on expected nuclear radiation dose rates in these two locations (see Figure A-14).

Environment 1 Inside Containment

Normal Operation

Temperature	80°F*
Pressure	14.4 to 14.7 psia*
Relative Humidity	25% in winter, 90% in summer*
Radiation	Figures A-7 through A-10

Accident Conditions Inside Reactor Containment

For BWR plants, the DOR Guidelines (Section 4) state that the environmental service conditions inside the drywell for the LOCA should be assumed to be 340°F for 6 hours. The design of the LaCrosse Boiling Water Reactor does not incorporate a drywell and has a relatively small reactor with a large containment; thus, at the September 11, 1980 site meeting with the NRC and Licensee, it was decided to use the LOCA pressure/temperature curve of Reference 7 for the equipment environmental qualification evaluation until the Licensee submits the revised profile to the NRC. As noted in Reference 29, however, the NRC has found the Licensee's MSLB analysis unacceptable because small break analysis, which could result in a higher temperature profile, was not performed. To complete this environmental equipment analysis, FRC used the Licensee's LOCA pressure/temperature profile because other analysis values were not available.

In addition, for plants equipped with automatic containment spray systems not subject to single component failure or delayed initiation, the Guidelines state that equipment qualified for the LOCA environment can also be considered to be qualified for the postulated MSLB. The design of the LaCrosse Boiling

*Dairyland Power Cooperative Letter LAC-6254, April 26, 1979.

Water Reactor does not satisfy this criterion. The LaCrosse studies do not establish a limiting condition temperature/pressure profile resulting from a complete spectrum of postulated break sizes, break locations, and single failures consequent to a MSLB accident inside the containment. Also, the design of the containment spray system is such that it is manually initiated, and the system is not safety-related. During the September 11, 1980 site meeting, the Licensee stated that calculations of the containment temperature and pressure profile are being performed and will be available for NRC review at a date to be determined later.

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

Temperature	See Figure A-1**
Pressure	See Figure A-1**
Radiation	<1 Mrd**
Relative Humidity	100% (assumed)
Flooded Level	663'8"
Chemical Spray	Demineralized water or river water

Environment 2 Turbine Building (640' elev. turbine condenser area, 654' elev. main stream piping area, and 629' elev. pipe tunnel)

Normal Operation

Temperature	70°F
Pressure	14.7 psia (assumed)
Relative Humidity	60-70% (assumed)
Radiation	Figures A-7 through A-9

Accident Conditions

Temperature	Figures A-11 through A-13+
Pressure	14.9 psia
Humidity	100% (assumed)
Radiation	Table A-1+
Flooded Level	3,283 cubic feet (equivalent elevation was not specified)+

**Dairyland Power Cooperative Letter LAC-5181, February 22, 1978.
+Dairyland Power Cooperative Letter LAC-7196, October 31, 1980.

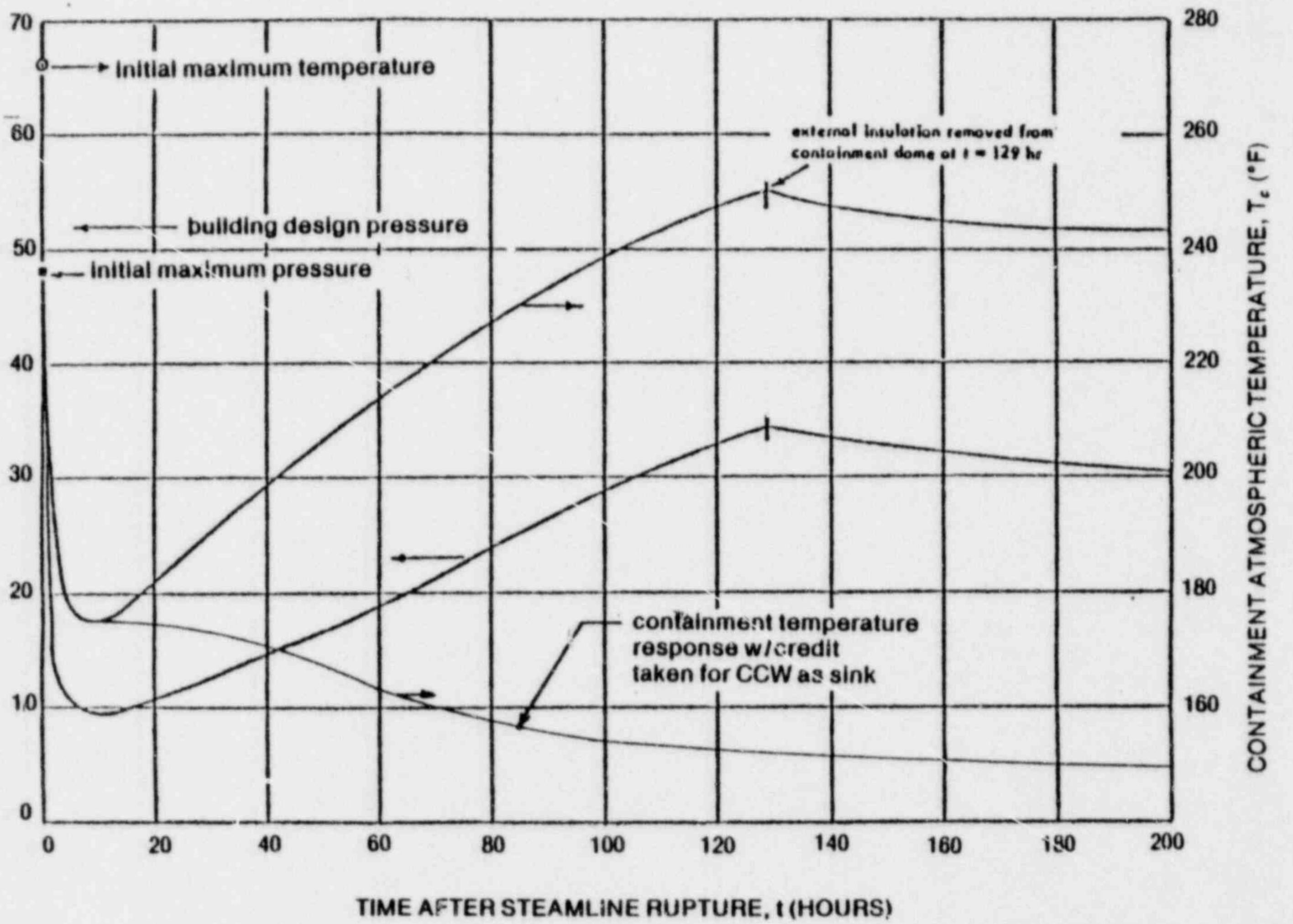
Table A-1

RADIATION LEVEL IN TURBINE BUILDING FOR LOCA/HELB IN CONTAINMENT

See Figures A-2 through A-5 for corresponding location of points in following table.

Location	Dose Rate (mRem/h) at T = 30 Minutes
1	3.6×10^3
2	5.0×10^1
3	1.0×10^4
4	2.5×10^3
5	3.3×10^4
6	3.0×10^3
7	1.1×10^4
8	3.4×10^3
9	4.8×10^4
10	2.8×10^4
11	1.1×10^4
12	1.6×10^6
13	2.43×10^7
14	1.61×10^6
15	1.61×10^6
16	1.66×20^6
17	Figure 5
18	Figure 5

ENVIRONMENTAL SERVICES CONDITIONS
TEMERATURE AND PRESSURE VERSUS TIME



LaCrosse Boiling Water Reactor

Figure A-1. Design Basis Accident Pressure and Temperature as a Function of Time

FIGURE SUPPLIED BY THE LICENSEE

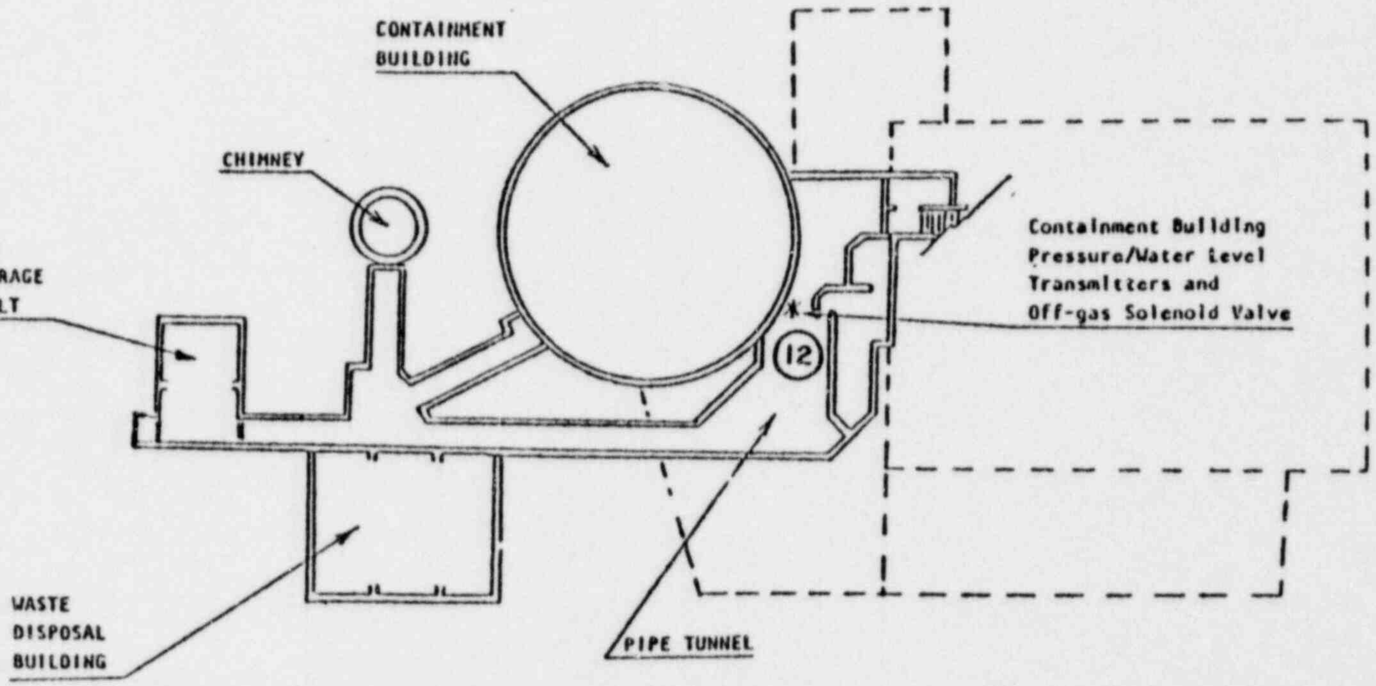


Figure A-2. LaCrosse Boiling Water Reactor, 629' Elevation Plant Layout

FIGURE SUPPLIED BY THE LICENSEE

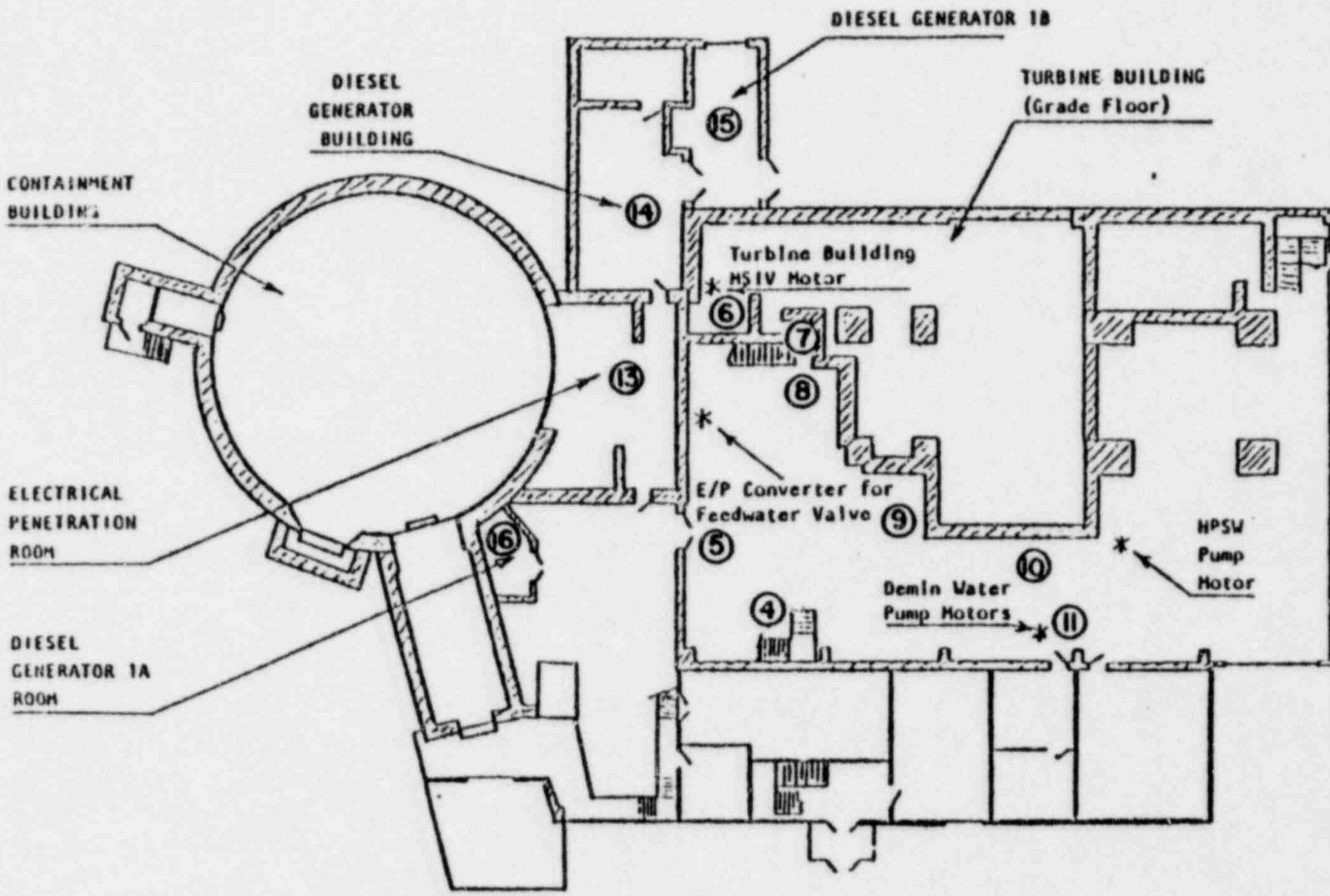


Figure A-3. LaCrosse Boiling Water Reactor, 640' Elevation Plant Layout

FIGURE SUPPLIED BY THE LICENSEE

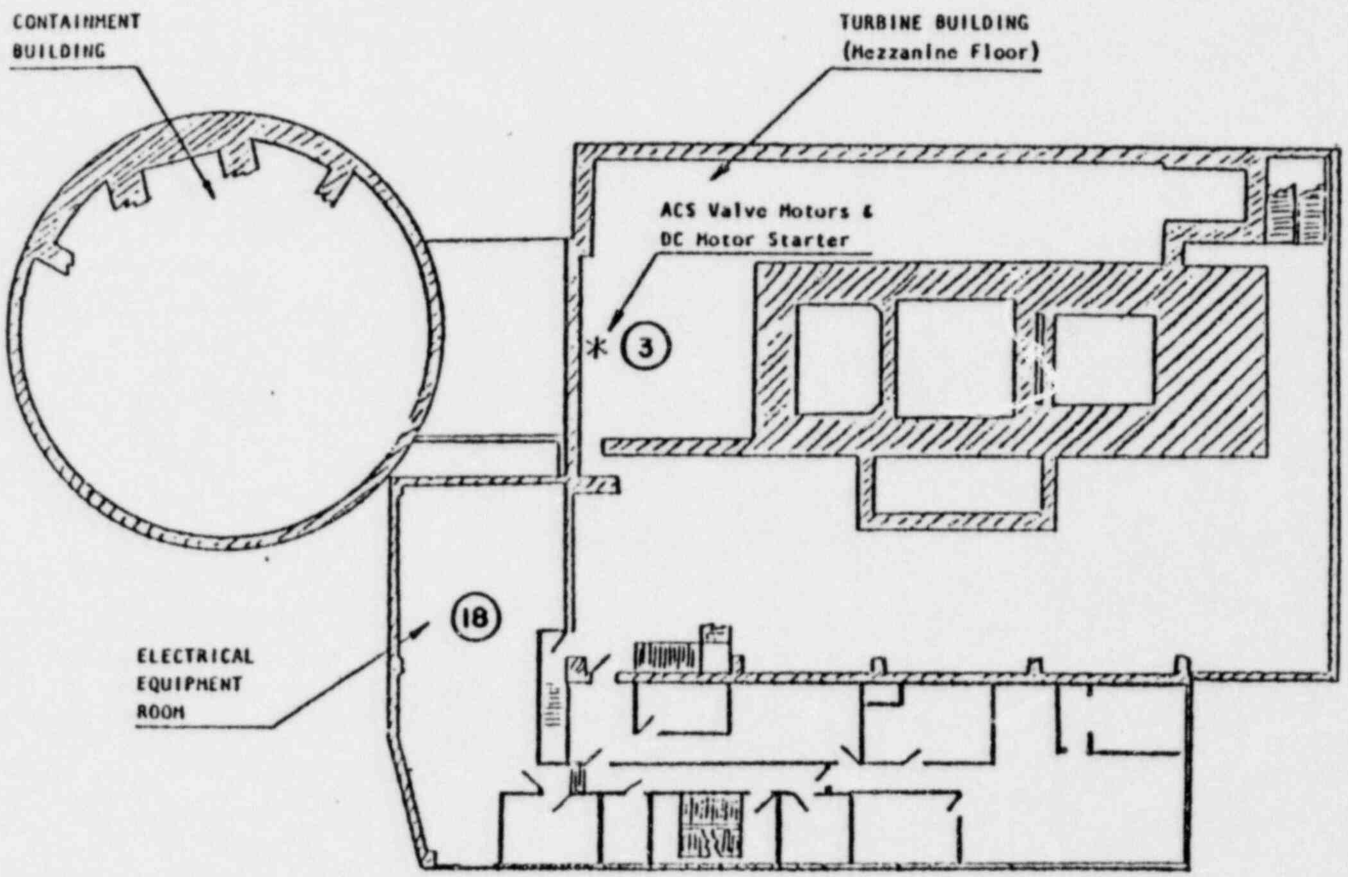


Figure A-4. LaCrosse Boiling Water Reactor, 654' Elevation Plant Layout

FIGURE SUPPLIED
BY THE LICENSEE

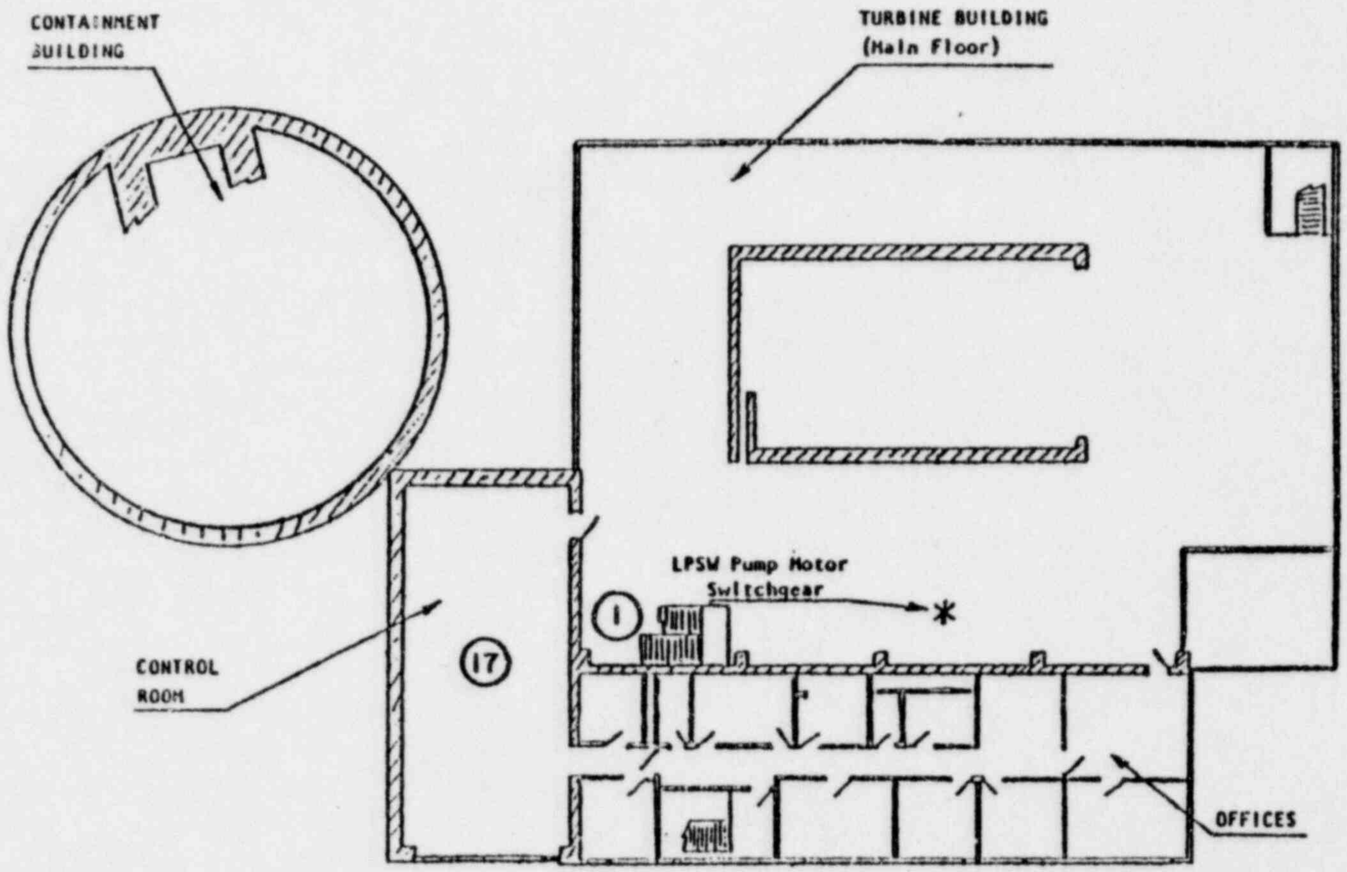


Figure A-5. LaCrosse Boiling Water Reactor, 668' Elevation Plant Layout

FIGURE SUPPLIED
BY THE LICENSEE

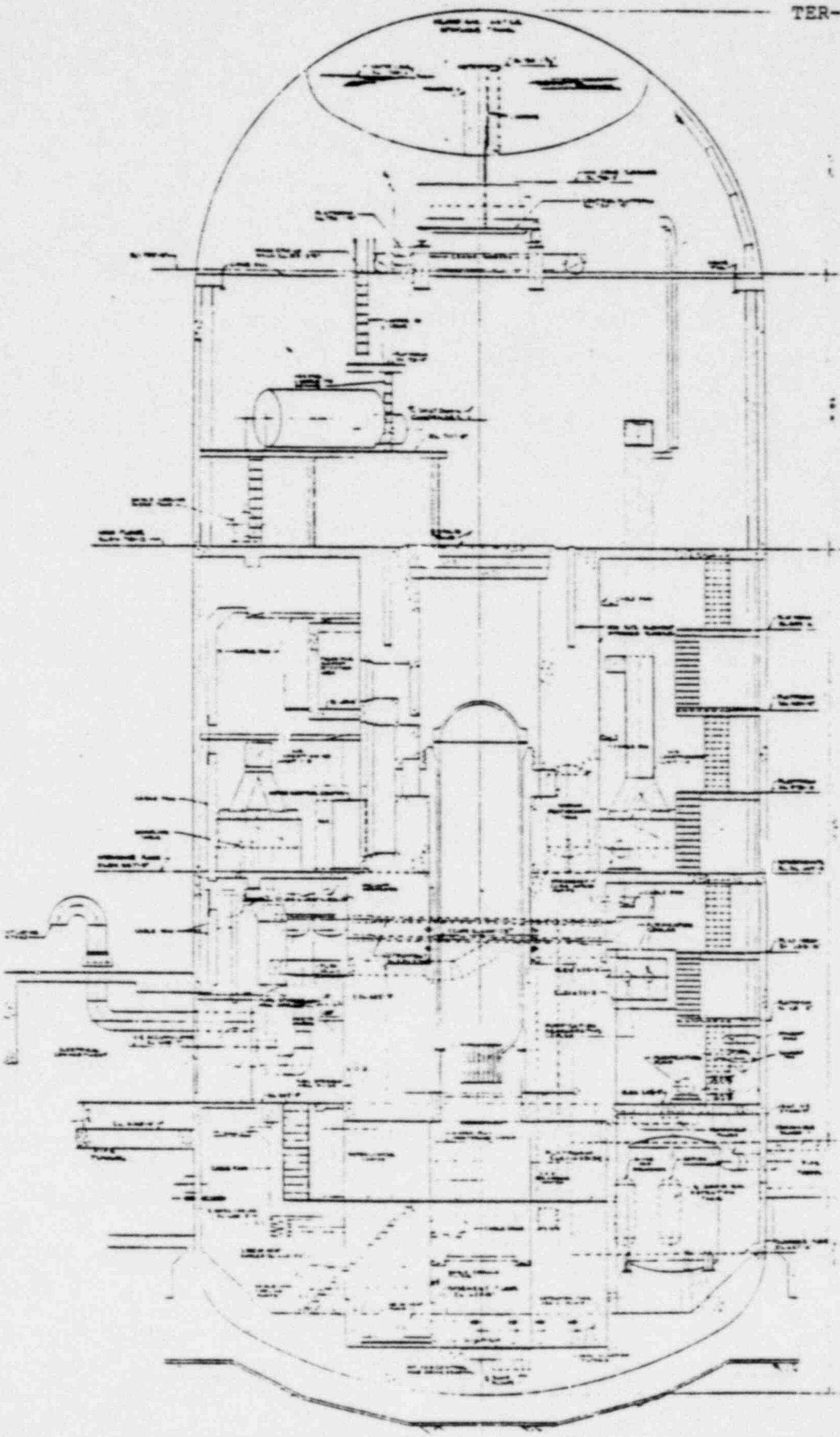


Figure A-6. LaCrosse Boiling Water Reactor Containment Section

FIGURE SUPPLIED BY THE LICENSEE

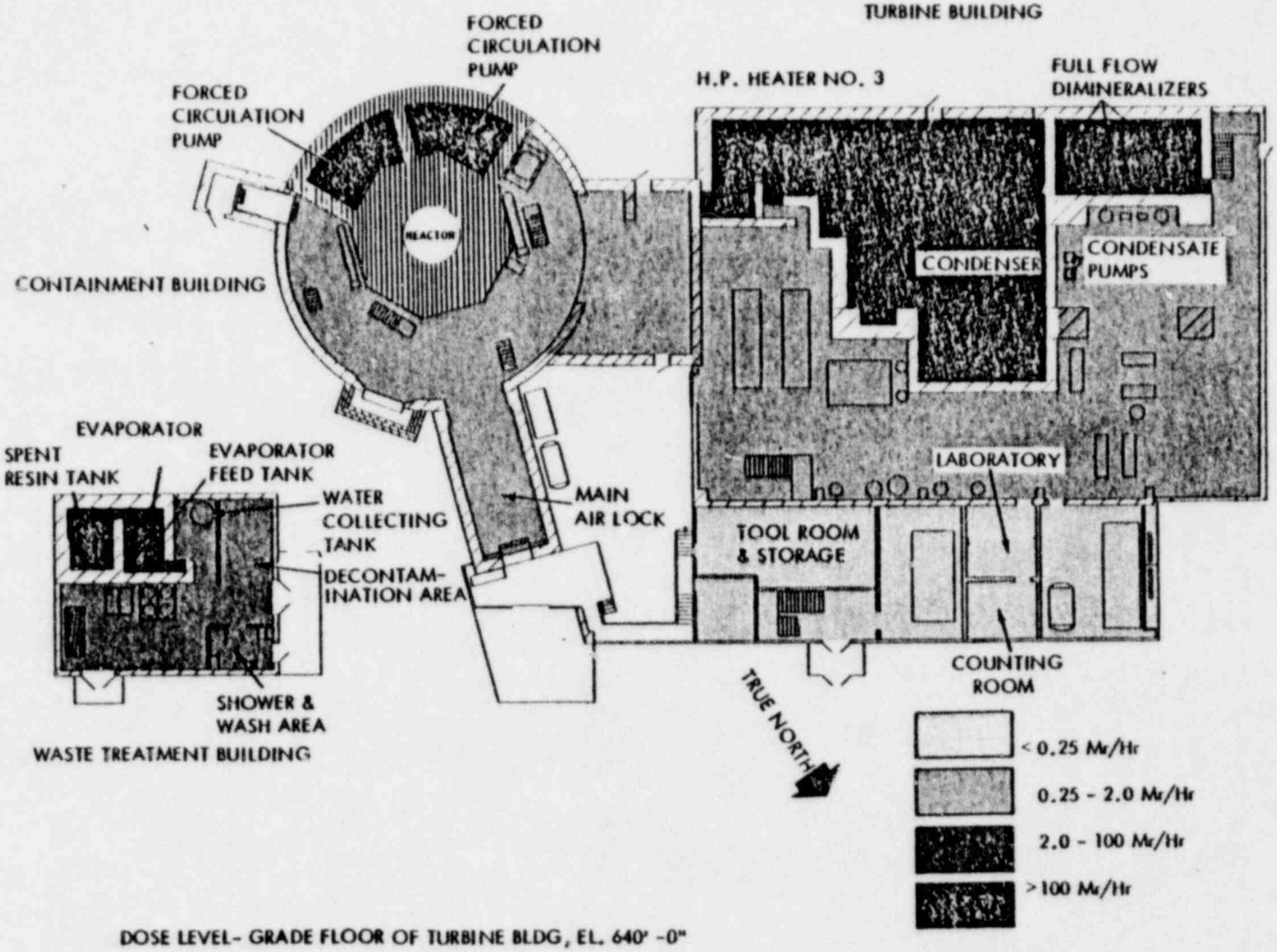
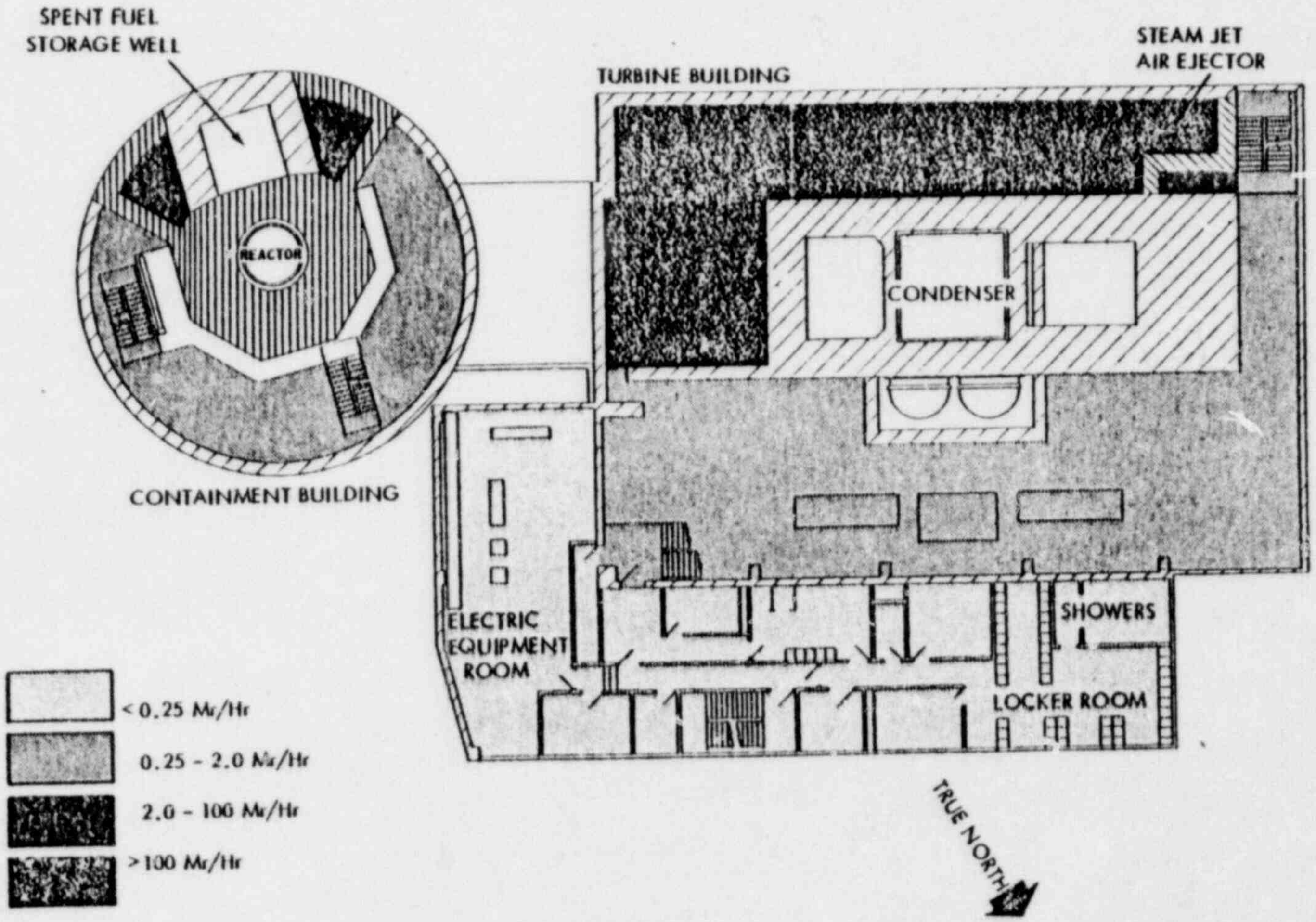


Figure A-7. LaCrosse Boiling Water Reactor Radiation Levels

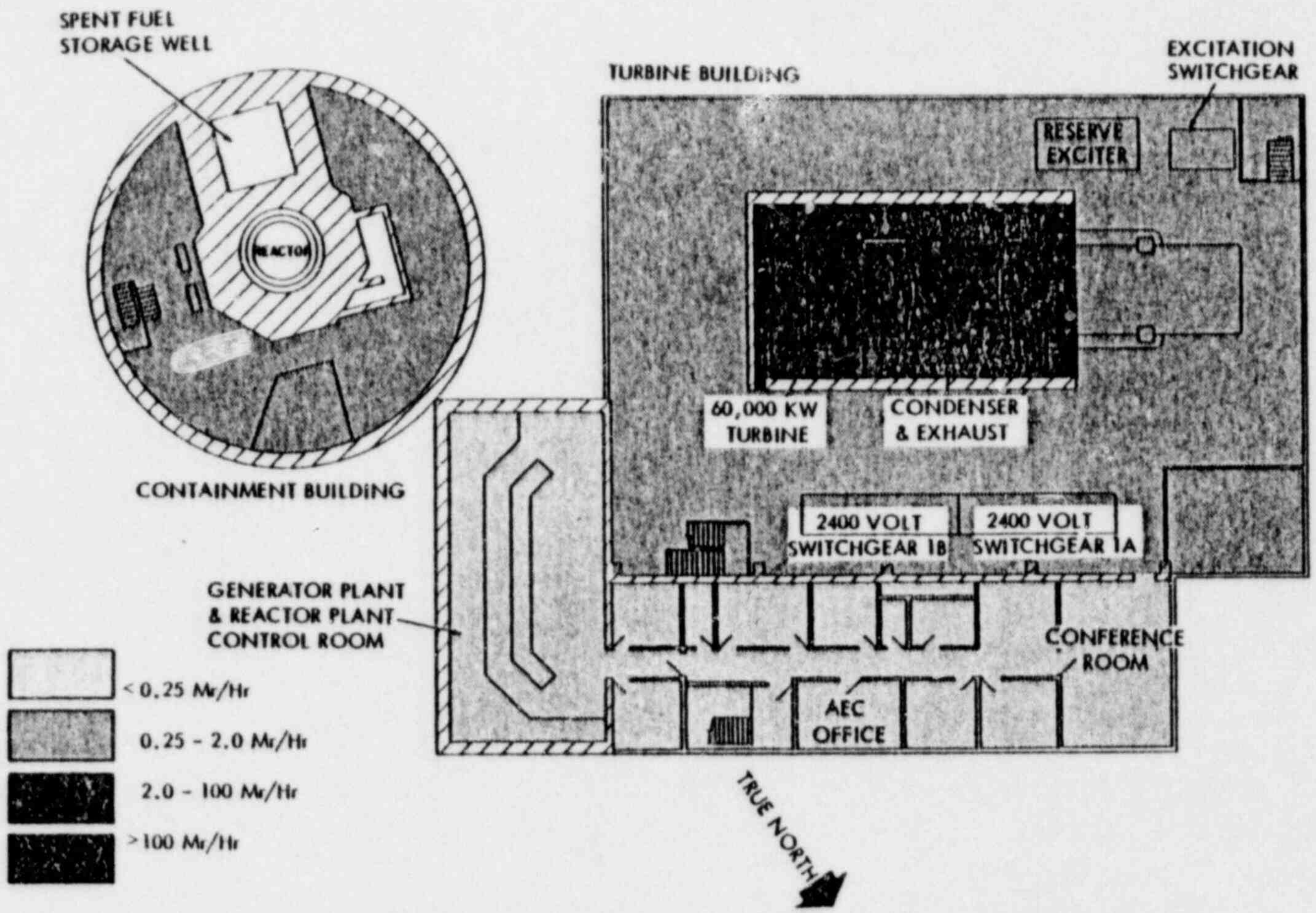
FIGURE SUPPLIED BY THE LICENSEE



DOSE LEVEL - MEZZANINE FLOOR OF TURBINE BLDG, EL. 654' -0"

Figure A-8. LaCrosse Boiling Water Reactor Radiation Levels

FIGURE SUPPLIED
BY THE LICENSEE



DOSE LEVEL - MAIN FLOOR OF TURBINE BLDG, EL. 668' -0"

Figure A-9. LaCrosse Boiling Water Reactor Radiation Levels

FIGURE SUPPLIED
BY THE LICENSEE

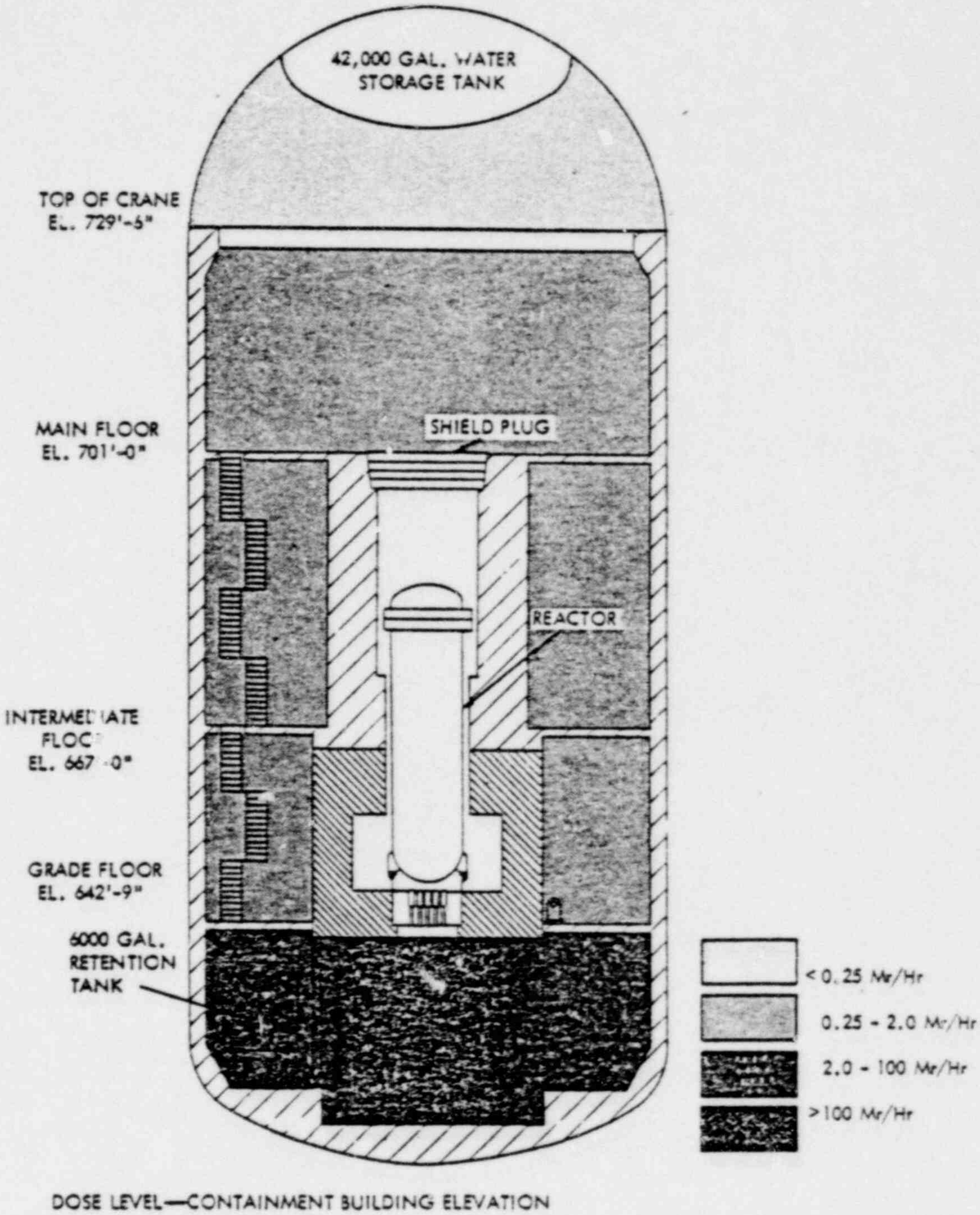


Figure A-10. LaCrosse Boiling Water Reactor Radiation Levels

FIGURE SUPPLIED
BY THE LICENSEE

Locations 1 & 2: Pipe Tunnel, Elev. 629'
ACS & MSIV Areas

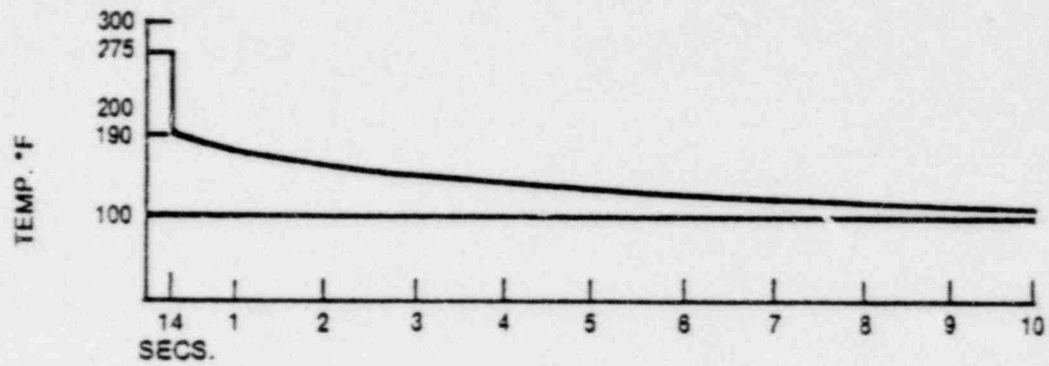


Figure A-11

Temperature Versus Time for Main Steam Line Break

Locations 1, 2: Pipe Tunnel, Elev. 629'
ACS & MSIV Areas

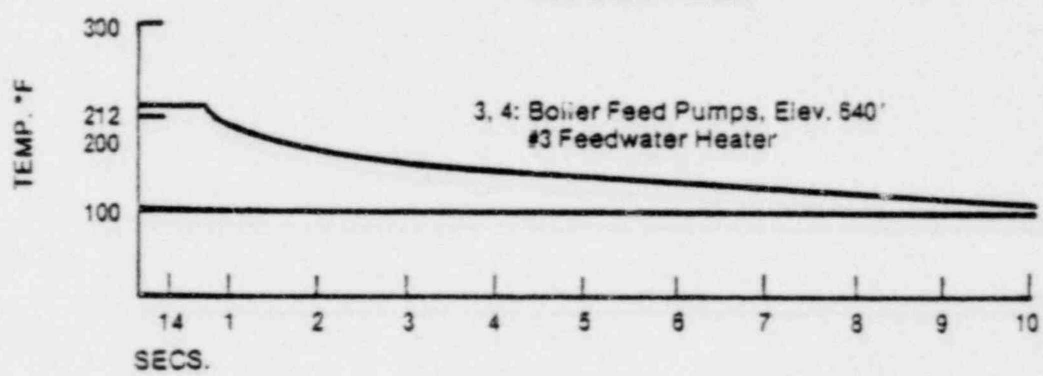


Figure A-12

Temperature Versus Time for Main Feedwater Line Break

FIGURE SUPPLIED
BY THE LICENSEE

Locations 3 & 4: Boiler Feed Pumps Elev. 640'
#3 Feedwater Heater

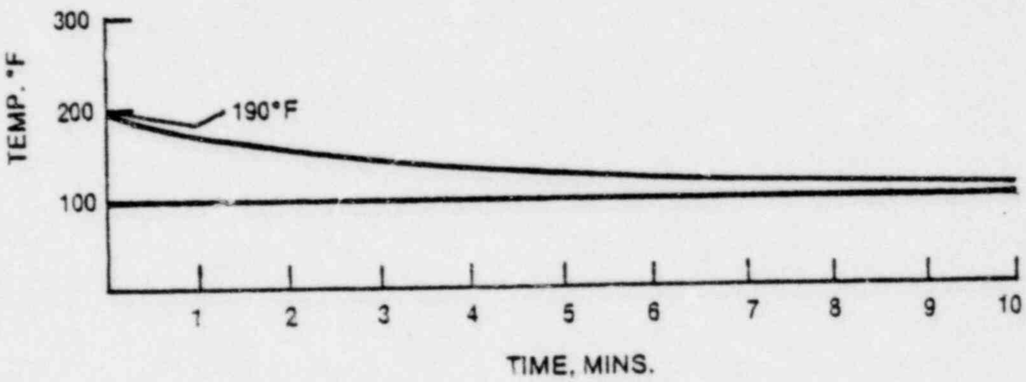


Figure A-13

Temperature Versus Time for Main Steam Line Break

FIGURE SUPPLIED
BY THE LICENSEE

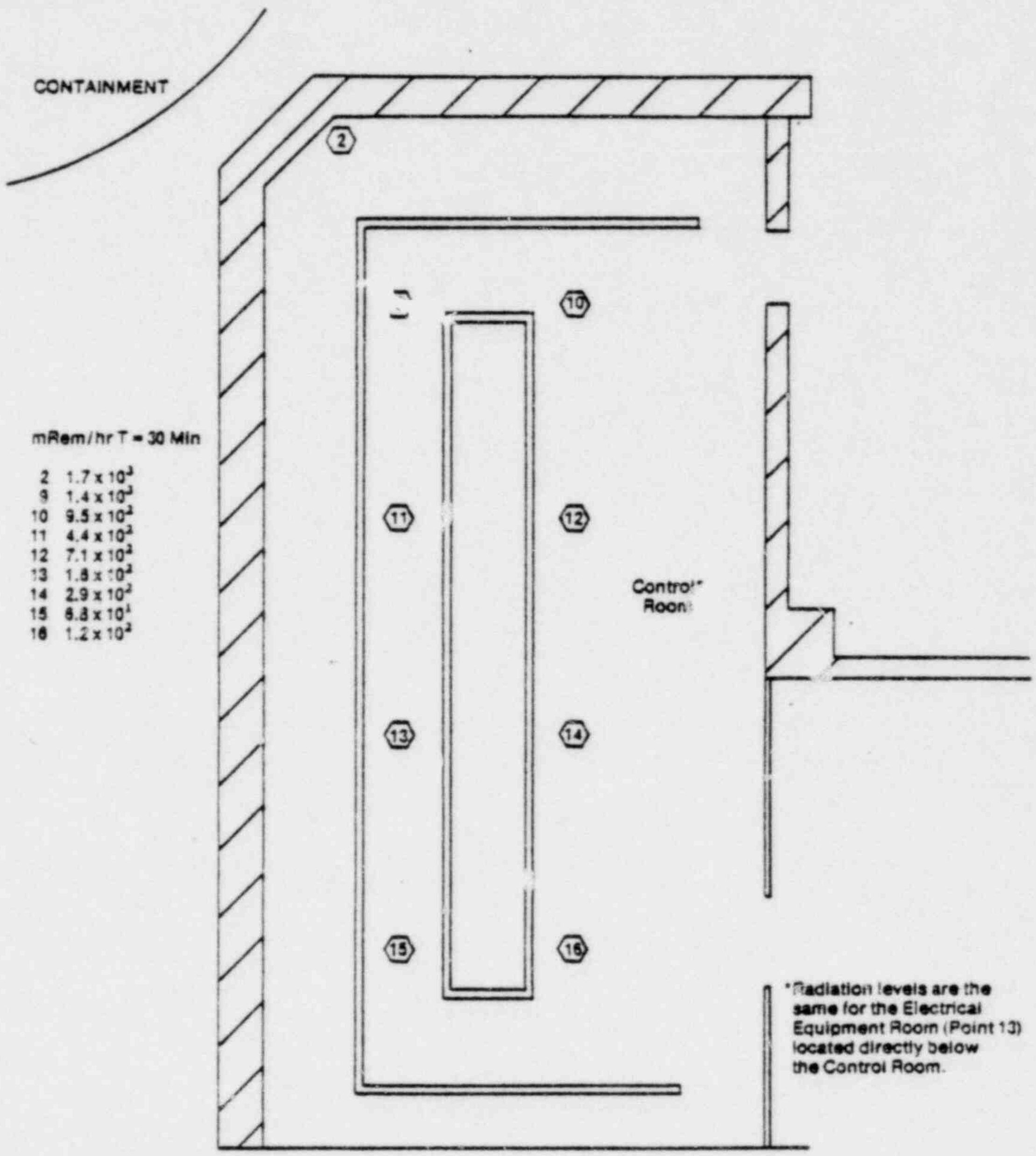


FIGURE SUPPLIED
BY THE LICENSEE

Figure A-14. Control Room Receiver Locations

APPENDIX B - LISTING OF SAFETY-RELATED ELECTRICAL EQUIPMENT

The following table lists the groupings of safety-related electrical equipment for the LaCrosse Boiling Water Reactor. Equipment item numbers provided in the table are used in the Equipment Environmental Qualification Summary Forms and in the equipment qualification discussions presented in Section 4.

This table was generated from the information provided by the Licensee [6, 7, 23], and shows manufacturer, model designation, plant location, time required, and qualification references.

PLANT NAME: LACROSSE

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	SUBMITTAL REFERENCE*	LOCATION	TIME REQUIRED	QUALIFICATION REFERENCES
1	Electric Motors Allis-Chalmers Type G with Class H Silicoflex Ins. (HPCS Pumps 1A and 1B)	2/1 (44)	Containment	4 h	None
2	Solenoid Valve ASCO WPX3815B3 (LPCS and MDS Vent Valves)	2/2, 5, 6 (1)	Containment	Long (Note 1)	None (Note 2)
3	Solenoid Valve ASCO HV202-301-4RG (HPSW Valve)	2/3 (2)	Containment	> 2.5 h	None (Notes 2,3,4)
4	Solenoid Valve ASCO HV202-924-4RG (Demin. Water Valve)	2/4 (3)	Containment	> 2.5 h	None (Notes 2,3,4)
5A	Solenoid Valves ASCO 8300B9RF (MDS and Demin. Water Valves	2/7-9 (Note 5)	Containment	0-3 sec (Note 1)	None (Note 4)
5B	Solenoid Valve ASCO 8300B9RF (Off-gas Vent Isol. Valve)	1/15 (4)	Pipe Tunnel	--	None (Note 3)
6	Solenoid Valves ASCO 8300B9F (HPSW Valves)	2/10 (9, 10)	Containment	0-3 sec	None (Note 4)

*Table No./Item No. from Reference 7 and/or (Item No. Used in Reference 23).

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	SUBMITTAL REFERENCE*	LOCATION	TIME REQUIRED	QUALIFICATION REFERENCES
7A	Level Transmitter Foxboro T/613DM-MS2-0 (Reactor Water Level)	2/11 (Note 6)	Containment	< 1 sec	None (Note 2)
7B	Level Transmitter Foxboro E13DM (Reactor Water Level)	2/11 (Note 6)	Containment	< 1 sec	TR-11 (Note 2)
8	Switchgear Allis-Chalmers Value Line MCC (Demin. Water Pump 1B)	2/12 (32)	Containment	--	None (Note 1)
9	Motorized Valve Actuators Limitorque Model Not Stated (ACS Pump Discharge Valves)	1/6 (Note 8)	Turbine Building	11 sec	13
10	Motor Starter Cutler Hammer K646676A	1/7 (34)	Turbine Building	11 sec	None
11	Electric Cable Mineral Insulation, Epoxy Sealant	(Note 7)	Containment	Long	9,17
12	Limit Switches NAMCO EA-180 (Reactor Steam Relief Valve)	(39)	Containment	--	18
13	Terminal Blocks Buchanan 218	(Note 9)	Containment	--	15,21
14A	Pressure Transmitter Foxboro T611-GM	1/12 (35)	Pipe Tunnel	11 sec	None

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	SUBMITTAL REFERENCE*	LOCATION	TIME REQUIRED	QUALIFICATION REFERENCES
14B	Pressure Transmitter Foxboro T631-2AS	1/12 (36)	Pipe Tunnel	11 sec	None
15	Level Transmitters Foxboro T613-DM	1/13 (37, 38)	Pipe Tunnel	20 sec	None
16	Motorized Valve Actuator Limitorque Model Not Stated (MSIV)	1/14 (Note 8)	Turbine Building	--	13
17	Electrical Penetrations Manufacturer and Model Not Stated	1/5 (42)	Containment	--	TR-8, TR-9
18	Solenoid Valve ASCO X-8344 (MSIV)	(11)	Containment	--	TR-6
19A	Solenoid Valve ASCO 8300B9RF (Reactor Cavity Vent)	(12)	Containment	--	TR-6
19B	Limit Switches Manufacturer and Model Not Stated (Reactor Cavity Vent)	(13)	Containment	--	TR-6
20A	Solenoid Valves ASCO LM931612 (Ventilation Inlet and Exhaust Isol. Valves)	(Note 10)	Containment	--	TR-6
20B	Solenoid Valves Barksdale 178350AC2A1 (Ventilation Inlet and Exhaust Isol. Valves)	(Note 11)	Containment	--	TR-6

ITEM NO.	EQUIPMENT ITEM DESCRIPTION	SUBMITTAL REFERENCE*	LOCATION	TIME REQUIRED	QUALIFICATION REFERENCES
21	Limit Switches Manufacturer and Model Not Stated (Ventilation Inlet and Exhaust Valves)	(Note 12)	Containment	--	None
22A	Solenoid Valves ASCO 8300B9RF (Reactor Off-gas Isolation Valve)	(Note 13)	Containment	--	TR-6
22B	Limit Switches Manufacturer and Model Not Stated (Reactor Off-gas Isol. Valve)	(Note 14)	Containment	--	None
23	Temperature Detectors Thermo Electric Ceramo J-116-G-304-00-20-1 (Containment Building Temperature)	(Note 16)	Containment	--	None
24	High-Range Radiation Monitor General Atomic RD-23 (Containment Building Radiation)	(Note 15)	Containment	--	None
25	Radiation Monitor Manufacturer and Model Not Stated (Containment Post-Accident Radiation)	(Note 17)	Containment	--	None

Note 1. Licensee notes that active functioning (de-energization to "safe" position) occurs within a few seconds.

Note 2. Licensee notes that ability to withstand pressure environment is demonstrated by fact that the equipment is operable after being subjected to the containment leakage rate test.

Note 3. Licensee notes that ability to withstand temperature of 350°F or greater was the equipment design requirement.

Note 4. Licensee notes that materials used have a radiation tolerance of 100 Mrd.

Note 5. In Reference 23, the Licensee added an additional valve that had been previously overlooked (No. 67-25-003). The applicable Reference 23 items are 5, 6, 7, and 8.

Note 6. In Reference 23, the Licensee corrected the model number for two of the three instruments. All three are covered by Licensee item 43.

Note 7. Equipment added by Licensee as separate item in Reference 6. In Reference 23, it is designated as item 33.

Note 8. This equipment was not addressed in Reference 23.

Note 9. Equipment added by Licensee as item 31 in Reference 23, after being discussed during site visit.

Note 10. Equipment added by Licensee as items 14, 15, 18, and 19 in Reference 23, after being discussed during the site visit.

Note 11. Equipment added by Licensee as items 16, 17, 20, and 21 in Reference 23, after being discussed during site visit.

Note 12. Equipment added by Licensee as items 22 through 25 in Reference 23, after being discussed during site visit.

Note 13. Equipment added by Licensee as items 26 and 27 in Reference 23, after being discussed during site visit.

Note 14. Equipment added by Licensee as item 28 in Reference 23, after being discussed during site visit.

Note 15. Equipment added by Licensee as items 29 and 30 in Reference 23, after being discussed during site visit.

Note 16. Equipment added by Licensee as item 40 in Reference 23, after being discussed during site visit.

Note 17. Equipment added by Licensee as item 41 in Reference 23, after being discussed during site visit.

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

The NRC transmitted to all SEP plants, Indian Point Units 2 and 3, and Zion Units 1 and 2 the DOR Guidelines for evaluating Class 1E equipment qualification and the "Guidelines for Identification of That Safety Equipment of SEP Operating Reactors for Which Environmental Qualification Is To Be Addressed." Based on these documents, the Licensee submitted a list of safety-related systems that must function in order to mitigate the consequences of a design basis accident [7]. Prior to the site visit, NRC developed an independent safety system list that was reviewed by the Licensee, NRC, and FRC representatives during meetings at the LACBWR site on September 11 and 12, 1980. As a result of these discussions, the following list represents systems and display instruments for which the Licensee and the NRC have determined that qualification is to be addressed.

Accident Mitigating and Safe Shutdown Systems

Reactor Protection and Safeguards Actuation+
Manual Depressurization+
Alternate Core Spray (diesel-driven pumps)+
MSIVs+
Emergency Power*
Containment Isolation+
Post-Accident Radiation Monitoring and Sampling+
High Pressure Core Spray+
Low Pressure Core Spray+

Accident Mitigating and Safe Shutdown Display Instruments

Reactor Water Level*
Reactor Steam Pressure*
Containment Pressure
Core Spray Flow
Containment Water Level
Containment Building Temperature
Containment High-Range Radiation Monitor

+Accident mitigating system.
*Required for both accident mitigation and safe shutdown.

APPENDIX D - EVALUATION OF PLANT AREAS NORMALLY MAINTAINED AT ROOM CONDITIONS

The DOR Guidelines state that safety-related equipment located in plant areas maintained at room conditions prior to, during, and after the design basis accident do not require environmental qualification because equipment failure can be expected to be random. Equipment is considered to be maintained at room conditions when its installed location is serviced by redundant HVAC systems powered from onsite, diesel-backed power sources. Room conditions are considered to be those for which industrial grade equipment is usually designed to operate (typically, 50 to 104°F).

Plant areas stated by the Licensee to be normally maintained at room conditions are:

Control room
Office areas
Electrical Equipment Room
Diesel Generator 1A Room
Diesel Generator Building
Crib House
Electrical Penetration Room
Turbine Building 668' elev., 654' elev., and 640' elev.
Diesel Generator Room 1B
Machine Shop.

FRC representatives have not reviewed the various ventilation systems servicing plant areas claimed to be maintained at room conditions in order to determine and verify compliance to the requirements of the DOR Guidelines. Review of equipment environmental qualification for equipment in these areas has been deferred at the Licensee's option (see Section 2.2.3 of this report).

APPENDIX E - INSPECTION OF CABLING

The NRC instructed FRC to select at random at least one component during the scheduled site visit and equipment inspection and to inspect all power feeders, cabling, junction boxes, and interlock devices associated with that component. This inspection was established by the NRC in order to spot-check the relative completeness of the Licensee's listing of safety-related equipment.

Because of time limitations at the time of the site visit, a complete walkdown could not be accomplished. It was noted, however, that the cabling in the containment was the mineral insulated type and that accessible safety-related detectors were located in sealed containers to protect against flooding.

APPENDIX F - LICENSEE ACTION ITEMS

The following is a list of action items that have resulted from the September 11, 1980 site meeting and subsequent conversations with the Licensee that require further review and resolution by the Licensee in order to adequately address the qualification issues for the Final Technical Evaluation Report.

ORIGINAL DITER ITEM

A. Completion of containment temperature/pressure profiles after 129 hours from a postulated LOCA are needed to better understand the heat removal mechanism and its long-term effects.

ADEQUACY OF RESPONSE:

This item was not completed by the Licensee and, therefore, the duration of the containment's high temperature and pressure values cannot be assessed. The length of time for which safety-related electrical equipment will be required to remain functional under these harsh conditions remains unknown.

This item remains open.

ORIGINAL DITER ITEM

B. Reactor vessel and containment vessel heat sink models and calculations are needed to support the temperature/pressure profiles listed in A.

ADEQUACY OF RESPONSE:

As discussed in Appendix G, Section G-1 and G-2, the Licensee presented an analysis of a component cooling water system which could be employed in the future as a containment long-term heat sink. However, the existing method of containment dome insulation removal followed by fire water spray on the dome has not been provided with the additional detailed documentation, which had been requested, in order to verify the adequacy of this cooling mechanism.

This item remains open.

ORIGINAL DITER ITEM

C. Review and resolution of conflicting items listed in Appendix G is needed.

ADEQUACY OF RESPONSE:

Four of the eight concerns mentioned in Appendix G remain unresolved.

ORIGINAL DITER ITEM

D. Environmental service conditions (temperature, pressure, time duration, and submergence) for HELB areas located outside containment are needed.

ADEQUACY OF RESPONSE:

This item has been resolved; however, detailed backup calculations have not been provided by the Licensee.

It should also be noted that the Licensee is expected to provide information after February 1, 1981 for plant areas normally maintained at room conditions in accordance with Section 2.2.3. The submittal should take into account all pertinent safety-related equipment and associated model numbers, normal environmental parameters, system operational modes, process and instrumentation diagrams, emergency power bus feeders, and other operating experience qualification information.

APPENDIX G - DOCUMENTATION CONCERNS

FRC has identified concerns or apparent document discrepancies and has recommended a solution. The Licensee initially provided summary information in response to these stated concerns. Where the answers were complete, the item was designated as closed. Further clarification or responses to the remaining concerns that were identified in the DITER were not made available by the Licensee except for the Licensee's suggested long-term heat sink: the component cooling water system. Although some heat sink calculations were provided for the proposed heat sink, several supporting details and the basis for the calculations were lacking.

CONCERN NO. 1: The initial temperature of the containment air used in the calculation may be higher or lower than the actual air temperature during normal operation.

In ACNP-66501, the Licensee presented a series of answers to questions dating back to a 1965 study of containment pressures due to such causes as a postulated metal-water reaction of the fuel cladding and supports, the possible burning of hydrogen formed from the metal-water reaction, and the buildup of containment pressure itself when the metal water reaction is neglected. A simplified equation is presented in which the initial temperature of the containment is assumed to be 80°F; however, other documents post this initial temperature slightly higher. The final temperature was calculated to be 272°F. Because these calculations indicate that the pressure approaches the design limit of the containment vessel, it is possible that not enough margin exists in relationship to the design limit of the containment vessel. The Licensee should determine if the plant's normal operating containment temperature is greater or less than 80°F, in order to establish that the final expected temperature and pressure inside containment are within design limits.

CONCLUSION:

The detailed calculations which were made in 1977 and initially referenced by the Licensee [21] were not made available by the Licensee in the October 31, 1980 submittal. As a result, FRC performed an independent review of the previous thermodynamic equation reference and was able to arrive at the same conclusion that was presented in the initial Licensee response. In order to arrive at this conclusion, however, it appeared that two basic assumptions had to be made: (i) that both the water and the steam being emitted from a pipe rupture are at the same temperature and (ii) that the emitting steam and the existing containment air are well mixed.

1. The assumption that water and steam are at the same temperature is both reasonable and conservative.

   Reasonable because practically all the water emerges from the reactor vessel, at least in the worst case.

   Conservative because assuming the water to be hotter than the steam (up to the saturation temperature that corresponds to total pressure = partial pressure (air) + partial pressure (steam) in well-mixed case) leads to lower calculated pressure and temperature in the containment.

2. The assumption that steam and air are well mixed is questionable and unconservative.

   Questionable because pockets of unmixed air, adiabatically compressed, might conceivable exist (a) inside equipment cabinets, and (b) above an overhead storage tank water pool, where air has entered via a manhole.

   Unconservative because (a) the pressure in the unmixed model is very slightly (>1 psi) higher, (b) the temperature of the steam is higher in the unmixed model, (c) the temperature of air is even higher than the steam. However, the probable main effect is a higher temperature in a pocket above water pool, which does not threaten equipment.

FRC concludes that even though the results of the calculation yield higher containment temperatures and slightly higher pressures when a more realistic nonhomogenous steam/air mixture is assumed, the containment design pressure margin does not appear to be in jeopardy, at least to the initial pressure and temperature excursion immediately following a LOCA or MSLB inside containment. This concern is resolved.

CONCERN NO. 2: The plant requires both a short-term and long-term cooling system to dissipate decay heat from the reactor and the containment vessel. It should be noted that Criterion 38 to Appendix A of 10CFR50 requires redundancy in components and features regarding containment heat removal.

Licensee document ACNP-66546 [32] concludes that a heat sink system should be provided to assure that the building design pressure is not exceeded for a period of indefinite duration following a major accident. This system would be properly classified as a necessary engineering safeguard, but no further description has been presented. In a follow-up telephone conversation, the Licensee stated that this proposed heat sink method has been deleted. The Licensee should identify which safety systems (both existing and proposed) are needed to provide for short- and long-term heat sink for the reactor vessel and the containment vessel. Backup documentation in the form of models with supporting calculations are needed to demonstrate that sufficient margins and backup systems are available to mitigate a potential accident.

CONCLUSION:

The Licensee submitted general calculations for the deployment of a future safety-related system, the component cooling water system, which apparently demonstrated that the operation of the system could provide a long-term heat sink for the containment building. The calculations that were presented did not show how the specific heat transfer values were determined, and no drawings of the system were made available which might portray routing location and system length. In addition, the calculations did not model the long-term heat sink for the reactor itself to demonstrate that decay heat was effectively removed without creating large temperature gradients. The Licensee should provide a qualified safety-related component cooling water system with detailed calculations or models which could verify the heat transfer coefficients for adequate long-term cooling as well as provide detailed calculations for the reactor vessel's heat sink.

Regarding the SAR method of long-term cooldown, the Licensee did not submit detailed analysis to support the assumptions made in the ACNP-66564 document that would allow total verification of the containment temperature and pressure profile curve shown in Figure A-1 of Appendix A. Calculations submitted to date provide a general equation for predicting the behavior of the containment for approximately 120 hours following a LOCA or MSLB. The long-term reduction of the temperature depends on dome insulation removal and dome water spray. The following detailed information would be needed in order to more adequately verify the general equation. DPC should:

1. Provide the basis for all heat transfer coefficients "h". (It is not sufficient to list a general bibliography.)

2. Review the reasonableness for assuming infinite diffusivity for a water pool heated from below, as well as the fact that some heating from above is also presumed.

3. Provide additional information relating to the calculation of the effect of spray on the dome and the overall effect of the overhead storage tank area.

   (a) Determine the basis for (and value of) the temperatures assumed on the outer containment surface due to spray with a fire hose, and establish the wet bulb temperature when the breeze is blowing. In addition, the fire hose water entrance temperatures, delta temperature rise due to the heat transfer, and the mass flow rate of the fire water should be determined.

   (b) Determine the heat transfer between dome and air/vapor pocket.

   (c) Determine if vapor is replenished by evaporation from the pool and what its driving forces are because, after vapor in the pocket has been condensed, migration of vapor up through the manhole may be negligible.

4. Provide the basis for the relative humidity and mass transfer coefficients of the containment air as well as their effect on the heat transfer coefficient at the pool/air and air/dome interfaces.

5. Provide further details and calculations concerning the heat flow below the pool, up through the pool, through the air under the dome, and up to and through the dome. In addition, the basis for any heat transfer or mass flow coefficients used should be made available.

CONCERN NO. 3: DPC should provide documentation that the automatic bus tie-in feature discussed in Reference TR-5 has been deleted. In Reference TR-5, the Licensee states that an automatic bus tie-in feature between a 120-V ac non-interruptible bus and a 120-V ac regulated bus was to have been deleted, but the necessary documentation to indicate that this has occurred has not been provided.

CONCLUSION:

FRC concludes that the concern has been resolved, because the Licensee has stated that Attachment 1 is a copy of the 10CFR50.59 safety analysis to Facility Change 75-12 which removed this feature.

CONCERN NO. 4: The primary concern is the implication that the feedwater system is required to perform a safety-related function; however, it has not been listed as a safety system. The secondary concern relates to the issue in Concern No. 2 regarding reactor heat sink provisions.

Chapter 14, page 31, of the FSAR discusses a major rupture-above-the-core type of accident. It states that the core can be partially uncovered and that the flashing of water within the core can be squelched by initiating emergency core spray by a low-low water level signal. Spray would then cool the core until it was covered again with water to prevent meltdown. The FSAR assumes the water level will be restored by the feedwater supply system with continued flow of at least 50 gpm of emergency cooling to keep the core covered. The apparent problem is that water cannot be continually pumped into the reactor, or to the containment, because of the eventual decrease in the containment air volume which would therefore increase the containment pressure. Also, the feedwater system has not been listed as a safety-related ECCS system. The Licensee should establish which safety system will be employed to keep the core covered.

CONCLUSION:

The emergency operating procedures should emphasize the use of safety-related systems in order to avoid operator confusion that has a high likelihood of developing if the operator has to rely on non-safety-related and unqualified instruments or systems to verify crucial parameters or provide a safety function.

The Licensee had responded that the high pressure core spray system is the safety system used to keep the core covered. Another safety system capable of the same function is the alternate core spray system using, if needed, the manual depressurization system.

By not stating that Emergency Operating Procedures would be reviewed and corrected, the Licensee has not addressed the concern. The Licensee should revise procedures accordingly or upgrade system and equipment to comply with the Emergency Operating Procedures so that misleading information from unqualified nonsafety systems will be minimized.

CONCERN NO. 5: Is the control room to be evacuated in case of a LOCA, and if it is evacuated what backup locations exist to safely shut down the plant?

Reference ACNP-66564 discusses permanent habitability of the control room with concern over personnel shift change, although another document--Reference 4, Attachment, LaCrosse Boiling Water Reactor Operating Manual--states in item 12 that it is possible for all personnel to be evacuated when the water lev in the containment building reaches 60 inches. The Licensee should clearly identify whether the control room or an alternate control location to be used during post-LOCA conditions.

CONCLUSION:

FRC concludes that the concern has been resolved, because the Licensee has stated that LACBWR procedures do not require an evaluation nor is any evaluation intended.

CONCERN NO. 6: The safety-related status of the demineralized water system has not been clearly defined.

Licensee Reference No. 6, page 3, describes the demineralized water pumps as not being required to function in the design basis event, and therefore not considered to be safeguard equipment; however, Licensee Reference No. 7, Table 1, Item 8 lists the motors for the demineralized water pumps as being safety-related. The Licensee should again review the safety-related status of both the demineralized water system and the high pressure service water system and formally identify the safety systems needed for the LOCA/HELB accident conditions.

CONCLUSION:

Conclusive evidence has not been provided to demonstrate that the long-term cooling method employing the alternate core spray system in conjunction with the manual depressurization system to dump heat to the containment is a viable heat removal scheme. It is not apparent that this method is as effective as one which would implement the shutdown condenser which, in turn, would be cooled by either the demineralized water system or the high pressure service water system. Without the backup documentation requested in Concern No. 2, there is no assurance that the long-term containment heat sink will perform as intended. The concern remains unresolved until evidence is submitted demonstrating that designated safety systems would remove heat from the core.

CONCERN NO. 7: Clarification of MSLB conditions in the turbine building was needed.

Reference No. 7, Note 7 to Table 1, states that a detailed analysis of the turbine building following a main steam line break was not made. Letter LAC-2935, dated January 23, 1975, in fact addresses the effect of postulated pipe failures outside the containment structure, and it specifically addresses the turbine building and has a graph depicting an expected pressure of 3 psi.

CONCLUSION:

FRC concludes that the concern has been resolved, because the references have been reviewed along with reconfirmation by a consultant. The turbine building peak pressure of 3.39 psig took into consideration that the structural integrity of the turbine building is maintained to contain the pressure. Turbine building design pressure is 0.17 psia, and therefore peak pressure would be limited to approximately that value because of rupture of the turbine building enclosure.

CONCERN NO. 8: The status of the HPSC system as a safety-related system requires further review.

In a September 26, 1980 telephone discussion, DPC stated that the high pressure core spray (HPCS) system was not needed to mitigate the effects of a postulated accident. It should be noted that Reference 7 of the Licensee's submittal contains a letter (LAC-2935) dealing with the consequences of the postulated pipe failure outside the containment structure; the letter states that the HPCS is the only cooling system which has sufficient reliability to ensure a safe shutdown for either an isolatable or non-isolatable pipe break. In addition, the HPCS system is capable of providing some long-term core cooling if the high pressure service water system can remain operational after a pipe break outside containment. It should be noted that since the HPCS system is backed up by the ACS system, there will be adequate short-term core cooling to ensure a safe reactor shutdown even if the high pressure service water system is damaged by a pipe break outside containment. Because the high pressure service water system is connected to the low pressure service water system, the latter would also be needed for long-term cooling. The Licensee should reevaluate what systems are designated as safety-related systems and have qualified components for LOCA and HELB accidents.

CONCLUSION:

The Licensee has stated that the high pressure core spray system and the alternate core spray/manual depressurization systems are safety-related. This concern is closed.

APPENDIX H - EVALUATION OF LICENSEE EXPLANATION OF THE ADEQUACY AND INTERIM OPERATION OF NON-QUALIFIED EQUIPMENT BASED ON SYSTEM OPERATIONAL CONSIDERATIONS

In the October 31, 1980 submittal from Dairyland Power Cooperative, the Licensee presented several system operational reasons for classifying unqualified components as satisfactory for either continued plant operation or interim operation until replaced by qualified components. The reasons include the availability of redundant systems (or components), time of operation, and involvement in specific design basis accidents.

At the NRC's request, FRC has evaluated these Licensee explanations. The results of these evaluations are presented in this appendix and referenced in appropriate sections of the text.

H.1 ALTERNATE CORE SPRAY/MANUAL DEPRESSURIZATION SYSTEM

LICENSEE POSITION:

> DPC's position with respect to post-accident core heat removal (emergency core cooling) has been that the high pressure core spray (HPCS) and low pressure core spray (LPCS) systems have been provided for short-term cooling only and can perform this function relying solely on a water inventory in the overhead storage tank. All long-term cooling is provided by the alternate core spray (ACS) system. Further, the Licensee has stated that the ACS system in combination with the manual depressurization system (MDS) provides a "totally redundant" backup to the LPCS system and that the ACS system is, independently, a suitable backup for the HPCS system.
>
> DPC has supported this position by stating that the ACS/MDS systems are "single-failure proof," and "were reviewed in 1976 and found to meet single-failure criteria and thus comply with the Interim Acceptance Criteria for ECCS."

FRC EVALUATION:

In order for equipment qualification to be deferred on the basis of the availability of a reliable (safety-related) backup system, the Licensee must demonstrate that such a backup system will perform its safety function when required. Information provided by the Licensee demonstrates only that the ACS/MDS system will continue to operate following a single-active failure within the system. Sufficient information has not been made available to FRC to allow an independent determination that the ACS/MDS system will perform its safety functions in the event of either a long-term passive failure or missiles, including those which could be generated by a high energy pipe break in non-safety systems. Despite this lack of evidence, certain credence must be given to the Licensee's claim that these systems are designed as safety-related systems which will perform their safety functions following a design basis event. Further, the scope of the present task does not include FRC's investigation of, or determination of compliance with, current criteria for engineered safety features.

FRC CONCLUSION:

Where the Licensee has invoked the availability of the ACS/MDS systems as justification for interim operation until a previously undocumented component is replaced, FRC has no technical objection.

In the case where the Licensee proposes to exempt from qualification certain components due to the availability of the ACS/MDS systems, FRC concludes that insufficient information has been provided. See TER References 4.6.1 and 4.6.2 for equipment discussions.

H.2 CONTAINMENT VESSEL OFFGAS VENT HEADER SYSTEM

LICENSEE POSITION:

> DPC's position regarding the containment vessel offgas vent header isolation valves and three-way vent routing valves has been that these components will not be required to operate in a post-LOCA environment. Specifically, DPC has indicated that the isolation valves will shut immediately upon a LOCA and will not require reopening, and that the vent header routing valve will not have to change position following a LOCA.

FRC EVALUATION:

The Licensee has determined that the containment vessel offgas vent header isolation valves and the three-way routing valves will perform their proper function immediately after a postulated accident and will not be required to change position later in the accident. FRC has no technical objection to the Licensee's position. The Licensee should review the possibility of using these valves at a later date for post-accident cleanup of airborne activity in order to confirm that long-term operation will definitely not be needed.

FRC CONCLUSION:

Because the Licensee has stated that these valves will not be required to function following an accident, FRC has no technical objection to the Licensee's justification for interim operation.

See TER Sections 4.7.3 and 4.6.13 for equipment discussion of the offgas vent header valves.

APPENDIX I - EVALUATIONS OF LICENSEE EXPLANATIONS OF EXEMPTION OF SPECIFIC EQUIPMENT ITEMS FROM QUALIFICATION REQUIREMENTS

This appendix contains the Licensee Responses and final FRC conclusions for equipment items FRC had listed in the DITER as possible safety-related equipment item oversights.

1. EQUIPMENT ITEM - OVERHEAD STORAGE TANK LEVEL INDICATOR

LICENSEE RESPONSE:

> Overhead Storage Tank Level Indicator - LACBWR does not have a remote level indicator on the overhead storage tank. It does have low and high level switches. Qualification of this equipment is not required, as the overhead storage tank is filled at all times when operating with a sufficient quantity of water to perform the short-term cooling function it was designed for. The known rate of water removal due to the use of positive displacement high pressure core spray pumps makes a qualified water level indicator unnecessary.

FRC CONCLUSION:

FRC has determined that the overhead storage tank level switch should be qualified for the containment environment. These switches perform their safety function, maintenance of the required water inventory in the overhead storage tank, prior to an accident and thus should be qualified for environmental conditions in the containment during normal operation. FRC does not agree that the use of positive displacement HPCS pumps makes the qualified water level indicator unnecessary.

As installed, these level switches appear, primarily, to control water level in the overhead storage tank at its technical specification level through operation of a demineralized water makeup isolation valve. As a secondary function, they provide indication. The Licensee's position has been that the overhead storage tank provides a water inventory for short-term cooling only and will not be refilled following an accident. As discussed in Appendix H, FRC has no technical objection to the Licensee position. Further, as the control system appears to be set up to maintain the water inventory at or near the design level, it is unlikely that the setpoint of the low level switch will provide any useful information about the time remaining before the operator must shift from the short-term (i.e., HPCS, LPCS) to the long-term (ACS) cooling mode.

FRC does not consider it acceptable to initiate a manual shiftover from short-term to long-term cooling on the basis of operating time of positive displacement pumps which may be run either singly or in combination. Proper timing of emergency core cooling shiftover is a significant issue. In more recent plants, this action is accomplished automatically on the basis of water storage tank level. If the shiftover operation is delayed, core cooling will be interrupted and the HPCS pumps will run dry. FRC feels that operator action should be supported by a qualified, low-level alarm which will provide adequate warning to the operator to initiate shiftover before the storage tank is empty.

2. EQUIPMENT ITEM - REACTOR CONTROL ROD DRIVE SCRAM SOLENOIDS

LICENSEE RESPONSE:

> Reactor Control Rod Drive Scram Solenoids - The design of the LACBWR control rod drive system includes a hydraulic scram system with dual solenoids which are redundant to each other. The removal of electrical power from either solenoid causes the associated control rod to scram to the fully inserted position in approximately 2 seconds. As the functioning of these control rod scram circuits would occur at the onset of a LOCA and these circuits do not have the capability of withdrawing the control rod once inserted, no qualification is required.

FRC CONCLUSION:

The Licensee has not provided suitable justification for these solenoids to be exempted from environmental qualification. These solenoids are exposed to normal operating radiation dose levels that may be significant from a material degradation standpoint.

FRC concludes that these solenoids should be qualified for the environment to which they are exposed.

3. EQUIPMENT ITEM - REACTOR OR MAIN STEAM PRESSURE TRANSMITTER

LICENSEE RESPONSE:

> Reactor or Main Steam Pressure Transmitter/Reactor Protection System Instrumentation - This system includes reactor water level (discussed as Item 43 in Enclosure 2), reactor power to flow instrumentation, and reactor primary system pressure. The power-to-flow and primary pressure were not designated as post-accident monitoring instruments and reactor water level, containment temperature, water level, and pressure are utilized for this purpose.

FRC CONCLUSION:

Figures 8.16 and 8.17 (Safety System Functional Block Diagram) of the LaCrosse Safeguards Report show the main steam pressure transmitter and the reactor pressure signal as providing input to the plant's safeguard system. Therefore, FRC concludes that the main steam and reactor pressure transmitters should be qualified for the environment to which they are exposed.

4. EQUIPMENT ITEM - MAIN STEAM FLOW TRANSMITTER

LICENSEE RESPONSE:

Main Steam Flow Transmitters - LACBWR is a single steam line BWR and would not utilize a steam flow transmitter following a LOCA to measure cooling through unaffected loops. As this is not a post-accident cooling mode and the steam line would be isolated, there is no requirement for qualified equipment.

FRC CONCLUSION:

FRC agrees with the Licensee's position that qualification is not required for this transmitter.

5. EQUIPMENT ITEM - REACTOR PROTECTION SYSTEM INSTRUMENTS

LICENSEE RESPONSE:

Reactor or Main Steam Pressure Transmitter/Reactor Protection System Instrumentation - This system includes reactor water level (discussed as Item 43 in Enclosure 2), reactor power to flow instrumentation, and reactor primary system pressure. The power-to-flow and primary pressure were not designated as post-accident monitoring instruments and reactor water level, containment temperature, water level, and pressure are utilized for this purpose.

FRC CONCLUSION:

The Licensee should provide the NRC with a list and associated electrical diagrams that would accurately identify all instruments in the LACBWR reactor protection system so that those instruments providing safe shutdown signals or control room indication may be reviewed for environmental qualification. The Licensee should document environmental qualification for any reactor protection system instruments that may previously have been overlooked, unless they are located in mild areas. For any electrical equipment in this system that has not been previously listed, maintenance surveillance analysis and design specification information, as well as the manufacturer and model number, should be provided in order to justify interim operation.

6. EQUIPMENT ITEM - SAFETY-RELATED CONTROL STATIONS

LICENSEE RESPONSE:

Any Safety-Related Control Station - The only safety-related control station at LACBWR is the reactor control room which is in a mild environment.

FRC CONCLUSION:

The Licensee is responsible for listing any safety-related control stations that are located in harsh areas and providing appropriate qualification documentation. The Licensee has determined that no such equipment exits in harsh areas and FRC, therefore, removes this item from the equipment list concerns.

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

EQUIPMENT ITEM NO.	DRAFT INTERIM TECHNICAL EVALUATION REPORT SECTION	FINAL TECHNICAL EVALUATION REPORT SECTION
1	3.3.3.1	4.6.1
2	3.3.2.1	4.6.2
3	3.3.2.1	4.6.2
4	3.3.2.1	4.6.2
5A	3.3.2.2	4.6.3
5B	3.3.2.2	4.7.3
6	3.3.2.2	4.6.3
7A	3.3.3.2	4.6.4
7B	3.3.3.2	4.5.2.1
8	3.3.3.3	4.6.5
9	3.2.1	4.3.3.3
10	3.3.3.4	4.6.6
11	3.3.2.3	4.3.3.1
12	3.3.2.4	4.3.3.2
13	3.3.3.15	4.5.2.3
14A	3.3.3.5	4.7.4
14B	3.3.3.5	4.7.4
15	3.3.3.11	4.7.5
16	3.2.1	4.3.3.3
17	3.3.2.6	4.2.1.1
18	3.3.2.6	4.6.7
19A	3.3.3.7	4.4.1
19B	3.3.3.7	4.4.1
20A	3.3.3.8	4.6.8
20B	3.3.3.8	4.6.8
21	3.3.3.9	4.6.8
22A	3.3.3.10	4.6.9
22B	3.3.3.10	4.6.9
23	3.3.3.12	4.5.2.2
24	3.3.3.13	4.7.1
25	3.3.3.14	4.7.2