NUREG-0991
Supplement No. 3

Safety Evaluation Report

related to the operation of
Limerick Generating Station,
Units 1 and 2

Docket Nos. 50-352 and 50-353

Philadelphia Electric Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

October 1984

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ABSTRACT

In August 1983 the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0991) regarding the application of the Philadelphia Electric Company (the applicant) for licenses to operate the Limerick Generating Station, Units 1 and 2 located on a site in Montgomery and Chester Countries, Pennsylvania.

Supplement 1 to NUREG-0991 was issued in December 1983 and addressed several outstanding issues. Supplement 1 also contains the comments made by the Advisory Committee on Reactor Safeguards in its report dated October 18, 1983. Supplement 2 was issued in October 1984 and addressed fourteen outstanding and fifty-three confirmatory issues and closed them out.

This Supplement 3 to NUREG-0991 addresses the remaining issues that require resolution before issuance of the operating license for Unit 1 and closes them out.

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APPENDICES

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

In August 1983, the Nuclear Regulatory Commission staff (hereinafter referred to as the NRC staff) issued its Safety Evaluation Report (NUREG-0991) regarding the application by the Philadelphia Electric Company (hereinafter referred to as the applicant) for licenses to operate the Limerick Generating Station, Units 1 and 2 (hereinafter referred to as Limerick or the facility), Docket Nos. 50-352 and 50-353. The Safety Evaluation Report was supplemented by Supplement No. 1 in December 1983 which documented the resolution of several outstanding issues in further support of the licensing activities and also contained the comments made by the Advisory Committee on Reactor Safeguards in its report dated October 18, 1983. In October 1984 Supplement 2 to NUREG-0991 was issued addressing and closing out numerous issues identified in the SER and in Supplement 1.

The purpose of this supplement is to further update the Safety Evaluation Report by addressing the remaining issues that require resolution prior to the issuance of an operating license for Limerick Unit 1.

Each of the following sections of this supplement is numbered the same as the corresponding section of the Safety Evaluation Report and Supplements No. 1 and 2. Each section is supplementary to and not in lieu of the discussion in the Safety Evaluation Report and Supplement No. 1 and 2 unless otherwise noted.

Copies of this SER Supplement are available for inspection at the NRC Public Document Room, 1717 H Street NW, Washington, DC and at the Public Document Room at the Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464. They may be ordered from the sources indicated on the inside front cover of this report.

The NRC Project Manager for Limerick is Mr. Robert E. Martin. Mr. Martin may be contacted by writing to the Division of Licensing, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Appendix A to this supplement is a continuation of the chronology of the staff's actions related to processing of the Limerick application.

This supplement to the Safety Evaluation Report was prepared by the NRC staff. The NRC members who were principal contributors to this report are identified in Appendix H.

1.8 Outstanding Issues

The SER identified certain outstanding issues. Supplements 1 and 2 to the SER reported the resolution of many of those issues. In this report, the staff discusses the resolution of all remaining items previously identified as open as well as additional information related to other sections of the SER. Issues previously reported closed are not addressed.

Issue	Section(s)	Status*
(1) emergency preparedness	2.3.3, 13.3	Closed (SSER-3)
(2) tornado-missile effects on ultimate heat sink	9.2.5	Closed (SSER-3)
(6) seismic/dynamic and environmental qualification of equipment	3.10, 3.11	Closed (SSER-2, 3)
(12) post-accident monitoring instrumentation	7.5.2.3	Closed (SSER-3)
(23) control room design review	18	Closed (SSER-3)
(25) failure modes and consequences of cooling towers	19	Closed (SSER-3)
(28) two-stage Target Rock Valves	3.9.3.4	Closed (SSER-3)
(29) pipe clamps	3.9.7	Closed (SSER-3)

1.9 Confirmatory Issues

The SER identified certain issues that have been essentially resolved to the staff's satisfaction but for which certain confirmatory information had not yet been developed. Supplements 1 and 2 reported the resolution of many of those issues. This report discusses the resolution of all remaining items previously identified as confirmatory. The list provided below updates the status of these confirmatory issues. Issues previously reported closed are not addressed.

Issue	Section(s)	Status*
(5) loading combinations, design transients, and stress limits	3.9.3.1	Closed (SSER-3)
(6) inservice testing of pumps and valves	3.9.6	Closed (SSER-3)
(9) overheating of gadolinia fuel pellets	4.2.3.2(4)	Closed (SSER-3)
(12) preservice inspection program	5.2.4.3, 6.6.3	Closed (SSER-3) Closed (SSER-3)
(22) fracture toughness of containment pressure boundary	6.2.7	Closed (SSER-3)
(26) instrumentation setpoints	7.2.2.1	Closed (SSER-3)
(30) restart of HPCI and RCIC on low water level	7.3.2.4	Closed (SSER-3)
(31) automatic switchover of RCIC	7.4.2.2	Closed (SSER-3)
(34) remote shutdown system	7.4.2.3	Closed (SSER-3)

Issue	Section(s)	Status*
(60) solidification/dewatering of solid waste (procedures)	11.4	Closed (SSER-3)
(63) assurance of proper ESF functioning (II.K.1.5)	15.9.3	Closed (SSER-3)

1.10 License Conditions

In Section 1.10 of the SER and Supplements 1, 2, and 3 the staff discusses issues for which a license condition may be desirable, unless satisfactory resolution was reached prior to licensing, to ensure that staff requirements are met during plant operation. The current status of these issues and the sections in which they are discussed are shown below.

License Condition	Status	Section
(1) turbine system maintenance program	License Condition	SER, 3.5.1.3
(2) fuel rod pressure limits	Resolved	SSER-3, 4.2.1.1
(3) Thermal hydraulic stability analysis for operation beyond Cycle 1	Resolved	SSER-3, 4.4.4
(4) scram system piping (NUREG-0803)	Confirmatory	SSER-3, 4.6
(5) addition of automatic isolation signals to RECW and CW isolation valves	License Condition	SER, 6.2.4.2 SSER-3, 6.2.4.2
(6) modifications to remote shutdown system	License Condition	SER, 7.1.4.4, 7.4.2.3 SSER-3, 7.4.2.3
(7) compliance with NUREG-0612 (Phase II, heavy loads)	License Condition	SER, 9.1.5
(8) shared emergency service water systems	Resolved	SSER-3, 9.2.1
(9) shared RHR service water systems	Resolved	SSER-3, 9.2.2
(10) shared control structure chilled water systems	Resolved	SSER-3, 9.2.8
(11) post accident sampling procedure (II.B.3)	License Condition	SSER-3, 14
(12) shared control structure ventilation systems	Resolved	SSER-3, 9.4
(13) personnel qualifications	License Condition	SSER-3, 13.1.2.2

License Condition	Status	Section
(14) implementation and maintenance of physical security plan	License Condition	SSER-2, 13.6
(15) prohibition of extended cycle operation with partial feedwater heating	License Condition	SER, 15.2
(16) addition of automatic isolation valves in hydrogen recombiner lines	License Condition	SER, 6.2.4.2 SSER-3, 6.2.4.2
(17) Exception to the schedular requirements of the Standard Review Plan for certain fire protection items	License Condition	SSER-2, 9.5.1
(18) ATWS Events (Generic Letter 83-28)	License Condition	SSER-2, 15.8
(19) Emergency response capabilities	License Condition	SSER-3, 7.5.2, 18
(20) Refueling floor connection to SGTS	License Condition	SSER-2, 6.2.3 SSER-3, 6.2.3
(21) Inservice testing of pumps and valves	License Condition	SSER-3, 3.9.6
(22) Environmental qualifications	License Condition	SER, SSER-2, 3.11
(23) Inservice inspection program	License Condition	SER, 5.2, 6.6
(24) Ultimate Heat Sink	License Condition	SSER-3, 9.2.5
(25) Emergency planning	License Condition	SSER-3, 13.3

2 SITE CHARACTERISTICS

2.3 Meteorology

2.3.3 Onsite Meteorological Measurements

In the SER, the staff concluded that the data recovery from the primary onsite meteorological tower (Weather Station No. 1) did not meet NRC criteria. This finding was based on review of meteorological data for a 5-year (1972-1976) period, during which the yearly data recovery ranged from 71 to 96% and the overall recovery was 84%. Also, the starting threshold of the anemometers did not meet the criteria recommended in RG 1.23. During the 5-year period of data record almost 18% of the hourly average winds were below the starting threshold of the anemometer, which means that wind direction cannot be defined during these periods.

The applicant completed installation of a new wind measuring system at the 9.1-m level on the primary tower before October 15, 1983. This system meets RG 1.23 criteria regarding starting threshold. Also, the applicant has submitted 6 months of meteorological data record (October 15, 1983 - April 15, 1984) from the primary tower. These data showed a valid joint data recovery of at least 96% for two vertical temperature difference (ΔT) measurements, and the wind measurements at each of the three elevations. For the 6-month period, hourly average winds below the starting threshold of the new wind measuring system occurred less than 5% of the time.

Therefore, the staff concludes that the meteorological instrumentation on the primary tower meets NRC criteria and that the applicant is showing progress towards providing acceptable data recovery through adequate maintenance.

The adequacy of the meteorological program regarding data recovery will be confirmed after receipt and evaluation of at least one year of data and review of the applicant's response to the staff's improvement recommendation in the Emergency Response Appraisal regarding meteorological instrumentation inspection procedures and documentation of the results of each inspection.

2.5 Geology and Seismology

2.5.2 Vibratory Ground Motion

Reg. Guide 1.60 recommends that the vertical response spectrum be 2/3 the horizontal response spectrum at low frequencies (less than .25 Hz) and equal to the horizontal spectra at high frequencies (greater than 3.5 Hz). At Limerick the vertical response spectra was assumed to be 2/3 of the horizontal spectrum at all frequencies.

Studies of western U.S. earthquakes (NUREG/CR-1175) have shown that the assumption that vertical ground motion levels are two-thirds those of horizontal motion is generally conservative. Consequently, the NRC staff has accepted that design response spectrum for vertical motion can usually be taken as two-thirds the horizontal response spectrum over the entire frequency range of interest for sites in the Western U.S. Recently, the staff was provided with the opportunity to examine the ratio of vertical to horizontal (V/H) motion recorded from eastern earthquakes. The records were obtained from aftershocks (magnitudes 4.0-4.8) of the New Brunswick earthquake of January 9, 1982 and from a magnitude 4.8 in New Hampshire which occurred on January 18, 1982.

The V/H ratio for these events was calculated from response spectra in the 0.2 to 30.0 Hz frequency range. The calculated V/H varied widely from site to site for a given earthquake as well as from earthquake to earthquake at a given site. In some cases for certain frequencies the ratio was as low as 0.1 while in other cases for certain frequencies it was as high as 2.1. No consistent pattern was observed. The average of the V/H data varied from 0.75 to 0.95 and is no simple function of frequency. Additional analysis by the staff as of this time, indicates that there are no systematic differences in earthquake source properties between the eastern and western U.S. Presently the very limited strong motion data set from the eastern U.S. is insufficient to draw generic conclusions with regard to differences in the V/H ratio between the eastern and western U.S. The most relevant information for Limerick is the data collected to generate the site specific spectrum used in estimating the horizontal ground motion for the site. The average V/H ratio for peak accelerations from these records (recorded in the western U.S. and Italy) is 0.65.

Based on our analysis we find the V/H ratio of 2/3 used at Limerick to be acceptable.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.9 Mechanical Systems and Components

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

In Section 3.9.3.1 of the SER, the staff stated that the applicant had committed to reconcile the final suppression pool hydrodynamic loads accepted by the staff with the loads used for plant design and document the results of the new loads adequacy evaluation in the FSAR.

In a letter dated August 8, 1984, the applicant provided the results of the reconciliation of the suppression pool hydrodynamic loads. The applicant stated that all safety-related BOP and NSSS piping components, equipment and their supports affected by the hydrodynamic load, both inside and outside containment have been included in the design assessment. Changes in the design such as additional supports, modification of existing supports or any other plant modifications as required to accommodate the suppression pool hydrodynamic loads have been completed. Detailed results of this assessment are documented in the plant Design Assessment Report (DAR) and the FSAR, Section 3.9.

Based on the results of the assessment performed by the applicant, the staff concludes that the applicant has demonstrated that all the affected safety-related piping components, equipment and their supports in the Limerick facility have been adequately designed to withstand the suppression pool hydrodynamic loads associated with the BWR Mark II containment design and other loads in combination as specified in the FSAR, Section 3.9.3.1. Therefore, the staff considers this confirmatory issue closed.

In Section 3.9.3.1 of the SER, the staff also addressed the issue of functional capability for NSSS essential systems. The staff requires that the functional capability of all piping components in essential ASME Code Class 1, 2, 3 piping systems designed to level C or D Service Limits be demonstrated. In FSAR, Section 3.9, Revision 27, the applicant stated that all ASME Class 1, 2, 3 NSSS essential piping systems are designed to meet the criteria described in the NRC staff-approved GE Topical Report NEDO-21985, "Functional Capability Criteria for Essential Mark II Piping," dated September 1978. Therefore, the staff considers this confirmatory issued closed.

In Section 3.9.3.1 of the SER, the staff stated that the applicant had committed to document the final results concerning the loading combinations, system operating transients and stress limits for the internal parts of the NSSS systems and components. Tables in Section 3.9.3.1 of the FSAR which contain this information have been completed. Based on a review of the information provided by the applicant, the staff has determined that the applicant's results meet the applicable design basis acceptance criteria described in FSAR Table 3.9.6, and, therefore, the staff considers this confirmatory issue closed.

3.9.3.4 Opening Pressure of Two-Stage Target Rock Safety/Relief Valves

In the Limerick SSER No. 1, the staff identified a generic concern regarding the opening pressure of two-stage Target Rock safety/relief valves (SRVs) which are to be used at Limerick Units 1 and 2. Experience at several other operating reactors has shown that these valves have had setpoints which drifted higher than the ±1% Technical Specification tolerance.

An extensive testing program was funded by the BWR Owners Group and was performed by General Electric Co., Target Rock Corp. and Wyle Laboratory. Several meetings have been held between these parties and the staff to discuss the exact nature of the high setpoint drift phenomena. The staff has also received a G.E. proprietary topical report - NEDE-30476 - from the Owners Group which identifies two major contributors to the upward setpoint drift: corrosive action which creates bonding between the pilot disk and seat and insufficient labyrinth seal clearance which creates friction on the pilot stem. The report recommends an improved maintenance and refurbishment procedure which is aimed at reducing pilot disk bonding and insuring greater labyrinth seal clearance.

At a meeting with the staff on November 10, 1983, the Owners Group discussed the conclusions and recommendations that are documented in NEDE-30476. The staff conclusion after the November 10 meeting was that the Owners Group recommended program would probably be sufficient to resolve the setpoint drift concern.

General Electric has incorporated the report recommendations for improved maintenance and refurbishment into supplement 14 of their Service Information Letter 196. For Limerick Unit 1 the applicant, in a letter dated June 22, 1984, stated that the Unit 1 Target Rock two-stage SRVs have been returned to Target Rock for implementation of all applicable supplements of SIL 196, including supplement 14.

The staff intends to expeditiously complete its final review of NEDE-30476 and related information being developed by one BWR licensee who is evaluating a change in SRV pilot disk material and a modification to the periodic inservice testing procedure for the SRVs. After completion of the review, the staff will publish its recommendations as to whether and by what means the provisions of SIL 196, Supplement 14 will be made mandatory and also whether any other actions are required to resolve this matter.

As stated above, the applicant has implemented the modification and refurbishment recommendations of all applicable supplements to G.E. SIL 196 for the Unit 1 SRVs. This adequately addresses SRV drift caused by the mechanism of friction in the valve labyrinth seal area. However, the staff has not concluded whether the SIL recommendations are sufficent to address setpoint drift resulting from pilot valve disk and seat corrosion. Nevertheless, the available two-stage valve data indicates that setpoint drift resulting from pilot disk and seat corrosion occurs less frequently than that caused by friction in the valve labyrinth seal area.

The staff expects to reach a generic resolution of the setpoint drift concern prior to shutdown of Limerick Unit 1 for its first refueling outage. The applicant has implemented all the applicable supplements of G.E. SIL 196 and, as stated in Section 5.2 of the FSAR, the applicant has installed considerably more SRV relieving capacity than required by the applicable edition of the ASME Code. Additionally, the Technical Specification surveillance requirement for testing frequency of the SRV set pressure has been made the same as the frequency of testing being performed by the majority of operating BWRs that utilize the Target Rock two-stage SRVs. This requires that at least fifty percent of the SRVs be tested at each refueling outage whereas the current ASME Code Section XI required frequency would require that about twenty percent of the valves be tested each time. This increased surveillance provides additional assurance that Limerick Unit 1 can be operated with no adverse effect on the health and safety of the public until the staff reaches a final generic solution on the matter of setpoint drift.

3.9.6 Inservice Testing of Pumps and Valves

In Section 3.9.6 of the SER, the staff stated that the results of its review of the issue of leak rate testing of pressure isolation valves would be reported in a supplement to the SER. There are several safety systems connected to reactor coolant system pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below reactor coolant system pressure. In order to protect these systems from reactor coolant system pressure, two or more isolation valves are placed in series to form the interface between the high-pressure reactor coolant system and the low-pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low-pressure systems, thus causing an intersystem loss-of-coolant accident. The Technical Specifications require that leak testing of pressure isolation valves be performed at periodic intervals and after all disturbances to the valve. The pressure isolation valves to be tested are listed in the Technical Specifications.

The applicant has agreed to categorize their pressure isolation valves for the core spray and residual heat removal systems as Category A or AC. These categorizations meet our requirements and we find them acceptable. Pressure isolation valves are required to be Category A or AC and to meet the appropriate valve leak rate test requirements of IWV-3420 of Section XI of the American Society of Mechanical Engineers Code except as discussed below. The allowable leakage rate shall not exceed 1 gallon per minute for each valve as stated in the Technical Specifications. The applicant has committed to test all pressure isolation valves to the 1 gallon per minute leak rate criteria.

In a letter dated September 4, 1984, the applicant has proposed to leak test at each periodic test interval as specified in Technical Specification section 4.4.3.2.2 and not each time the valve is disturbed for those systems which are rated at a lower pressure than the reactor coolant system. We find this acceptable for the following reasons: (1) full closure of these valves is verified in the control room by direct monitoring position indicators, (2) inadvertent opening of these valves is prevented by interlocks which require the primary system pressure to be below subsystem design pressure prior to opening, and (3) gross intersystem leakages into the core spray will be sensed and alarmed in the control room. (4) Before the first refueling outage, the residual heat removal pump discharge line pressure for each of the four RHR pumps will be observed and recorded once per shift from indicators in the auxiliary equipment room. This will inform the operators of any pressure increase due to leakage past the following valves - HV-51-1F041 A, B, C, D, HV-51-142 A, B, C,

D, HV-51-1F017 A, B, C, D, HV-51-1F050 A, B, HV-51-151 A, B, HV-51-1F015 A, B, HV-51-1F009, HV-51-1F008. Prior to startup after the first refueling outage, pressure monitors with alarms in the control room will be installed to detect leakage past these valves. (5) Gross intersystem leakage into the core spray and residual heat removal system may also be detected by monitoring the narrow range suppression pool level instrumentation and by monitoring flow to the radwaste collection system.

Based on a review of the information provided by the applicant, we conclude that the applicant's commitments to periodic leak testing of pressure isolation valves between the reactor coolant system and low-pressure systems will provide reasonable assurance that the design pressure of the low-pressure systems will not be exceeded, thus reducing, the probability of an occurrence of an intersystem loss-of-coolant accident, and is acceptable. Therefore, the staff considers this confirmatory issue to be closed. However, prior to startup after the first refueling outage we will require that the applicant complete the design modification to the residual heat removal pump discharge line monitor as described above.

In the SER the staff stated in Section 3.9.6 that the relief the applicant requested from the pump and valve testing requirements of 10 CFR 50.55(a)(g) is warranted for a portion of the initial 120-month period during which the staff completes its detailed review. By letter dated June 15, 1984, the applicant submitted revision 4 of the Limerick 1 inservice testing program for pumps and valves. The staff has also reviewed this revision to the program and finds that the type of relief previously granted in the SER is also warranted for revision 4 for the same reasons as stated in the SER.

3.9.7 Evaluation of Allegations Regarding Class 1 Piping Design Deficiencies

In Section 3.9.7 of the SER Supplement No. 1, the staff addressed the issue of the stiff pipe clamp. The applicant was requested to provide information regarding stiff pipe clamps as described in IE Information Notice 83-80. In a letter dated August 8, 1984, the applicant responded to the staff's concern. The applicant has provided a list of E-System clamps including clamp locations for BOP and NSSS piping systems. The applicant stated that stress evaluations to consider clamp induced stresses for E-system clamps located at or near elbow welds have been completed for BOP piping systems. The evaluation results showed that piping stresses are within the applicable code allowables. The applicant also stated that these results concur with investigations by both General Electric Company and Bechtel's Corporation which indicated that stiff pipe clamps do not cause stresses or fatigue levels higher than the governing stresses or fatigue levels in these piping systems.

With respect to the post-installation control of the clamp preload, the applicant stated that preload requirements for the E-System clamp installation are controlled by specification 8031-p-143-30-7. This specification is also used to control post installation preload of the E-System Clamps.

Based on a review of the information provided by the applicant, the staff determined that the applicant has provided reasonable assurance that the effects of the clamp-induced pipe loadings have been adequately considered in the Limerick piping design and, therefore, the staff considers this issue to be closed.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

3.10.1 Seismic and Dynamic Qualification

The staff's evaluation of the applicant's program for qualification of safety-related electrical and mechanical equipment for seismic and dynamic loads consist of: (1) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general, and (2) an audit of the selected equipment items to develop the basis for the staff judgment on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program. The Seismic Qualification Review Team (SQRT) consists of engineers from the Equipment Qualification Branch (EQB) and the Idaho National Engineering Laboratory (INEL, EG&G). The SQRT has reviewed the equipment dynamic qualification information contained in the pertinent Final Safety Analysis Report (FSAR) Sections 3.9.2, 3.9.3 and 3.10 and made a plant site audit on January 17 through January 20, 1984 to determine the extent to which the qualification of equipment, as installed at Limerick 1, meets the current licensing criteria as described in Regulatory Guides 1.100 and 1.92, Standard Review Plan (SRP) Section 3.10, and Institute of Electrical and Electronics Engineers' IEEE 344-1975 standards. Conformance with these criteria are required to satisfy the applicable portions of the General Design Criteria in 1, 2, 4, 14, and 30 of Appendix A to 10 CFR Part 50, as well as Appendix B to 10 CFR Part 50 and Appendix A to 10 CFR Part 100.

Based on the review of the methodology and procedures of the equipment seismic and dynamic qualification program contained in the FSAR, the meeting of March 4, 1983 with the staff, and the letter of March 31, 1983, the applicant was requested to provide additional information on his assertion that the Limerick Safety Relief Valve (SRV) discharge loadings have low frequency content, around 6 to 10 Hz, and that fatigue failures as a result of normal plant loads, including SRV actuation loads, were not a concern. In his response of October 25, 1983 the applicant stated that the Limerick SRV discharge in the suppression pool through "T"-quenchers which was developed in conjunction with Susquehanna Steam Electric Station (SSES) by Kraftwork Union (KWU). KWU developed the specification for SRV discharge load in the suppression pool. The basis of the applicant's conclusion is the work done by KWU, NUREG-0802, Safety/Relief Valve Quencher Loads: Evaluation of BWR Mark II and III Containments, and review of power spectral density (PSD) from the Karlstein test. It is, however, recognized that some high frequency content (40 Hz) does exist and according to the applicant has been included in load combinations used for equipment qualification.

With respect to the fatigue issue the applicant indicated in the letter of March 31, 1983 that he had reviewed fatigue effect analyses performed to date by other utilities, and loads at Limerick that result from SRV actuations. The review indicated that fatigue failures as a result of normal plant loads, including SRV actuation, were not of concern. However, based on the results of the meeting and discussions and to address the issue on a plant-specific basis, the following additional actions were taken by the applicant:

(1) Conduct extended duration testing of equipment that were yet to be qualified including components of the anticipated transient without scram/scram discharge volume (ATWS/SDV) modification package.

(2) Perform an analysis on one piece of equipment each in the reactor building and containment structures to further demonstrate that for Limerick the fatigue usage factors are less than 1.0.

According to the applicant's submittal of November 7, 1983, a number of pieces of Limerick Generating Station (LGS) equipment were qualified by extended duration testing. A brief summary follows: There were 17 pieces of equipment which underwent extended testing. A typical test on Westinghouse 250V DC Motor Control Center consisted of five OBE (Operating Basis Earthquake), one SSE (Safe Shutdown Earthquake), and one worst case RRS (Required Response Spectra) in addition to 20 minutes duration testing in both biaxial orientations for envelope of SRV and LOCA spectra. There were 13 pieces included in the ASTW/SDV Modification Equipment program. These were subject to SRV vibratory aging for 15 minutes in each horizontal/vertical orientation. The results in each case were satisfactory.

The applicant analyzed accumulator tanks, located inside the containment, and HVAC Panels, located in the reactor/control building. These pieces were selected for the fatigue evaluation because the stresses due to worst case loading were relatively high. The analyses were done per ASME B&P Vessel Code Section III, Subsection NB, 1983 Edition for cumulative usage factors due to fatigue effects. The usage factors for bolts, shell, clamp beam and welds for the tanks as well as the angle sections, plates, attachment bolts, and partial penetration welds for the panel were shown to be less than 1.0.

For the audit portion of the staff's evaluation process a representative sample of safety-related electrical and mechanical equipment, as well as instrumentation, included in both NSSS and BOP scopes, was selected. The plant site audit consisted of field observations of the final equipment configuration and its installation. This was immediately followed by the review of the corresponding test and/or analysis documents which the applicant maintains in their central files. Observation of the field installation of the equipment is required in order to verify and validate equipment modeling employed in the qualification program.

The SQRT review of the Limerick 1 Generating Station identified the following concerns relating to the seismic and dynamic qualification of only one equipment item, for which the applicant was requested to provide additional information for the staff review and acceptance.

Equipment Specific Issues - RCIC Steam Turbine Assembly

(1) The turbine governor and electrical accessories as originally installed at Limerick Unit 1, must be upgraded to be similar to the turbine which was tested.

(2) There were two qualification tests performed. In the first test program, #8 taper pins were used for coupling-end alignment. One of these pins failed after an accumulated test time of about 15 minutes. The turbine for the second test program, which was a success, used #9 taper pins. It also had lock plates for pedestal bolting. Thus, the Limerick Unit 1 turbine must be upgraded with #9 taper pins and lock plates for the pedestal bolting.

(3) The existing trip and throttle (T&T) valve in the Limerick Unit 1 turbine has a General Electric S&K trip solenoid (push to trip), whereas the successfully tested turbine used a Thrombetta trip solenoid (pull to trip). During the first qualification test program, with Thrombetta assembly, it became necessary to increase spring stiffness to 25 lb/inch in order to prevent trip latch separation during the resonance search tests. The T&T valve solenoid should be replaced with Thrombetta assembly and stiffness checked or justification provided.

(4) In the first test program, it was evident that structural improvements were required in the turbine auxiliary piping. These were implemented in the second test program which was successful. However, each turbine installation has somewhat of a unique piping arrangement. For the turbine oil piping adequacy, therefore, the Limerick Unit 1 as installed piping should be reviewed and adequate supports provided.

(5) The qualified life and resulting preventive or replacement schedule for the new accessories should be incorporated in the maintenance manual.

In the letter of August 1, 1984, the applicant confirmed that the above upgrading and modifications to the Limerick Unit 1 RCIC turbine assembly have been completed except for the installation of threaded taper pins which assist in maintaining alignment after pedestal bolting. The hold down bolts which attach the turbine pedestal to the baseplate have been installed. General Electric Company and the turbine assembly vendor, however, require that taper pins be installed after final (hot) alignment of the turbine assembly. Hot alignment is scheduled after nuclear steam has been applied to the turbine assembly, approximately 6-12 weeks, based on GE power ascension schedule, after fuel load, to bring it up to operating temperature and pressure. The schedule for taper pin installation on the Limerick Unit 1 HPCI turbine assembly is the same as described above. In the letter of September 10, 1984, the applicant further stated that RCIC and HPCI turbines are not required to be operable when the system pressure is less than 150 psi. Prior to reaching this pressure level, the taper pins will be inserted in place according to the requirement of Terry turbine instruction manual. This is acceptable to the staff.

Based on the above, the staff concurs that with the exception of the installation of the threaded taper pins, the modified assembly is now similar to the turbine which was used for dynamic qualifications testing, thereby achieving qualification of the Limerick Unit 1 RCIC turbine assembly. The staff will confirm that the taper pins are properly installed. This open item is closed.

Justification for Interim Operation

Only one category of equipment, the residual heat removal service water process radiation monitor (RHRSW PRM), for which qualification is not expected to be fully completed by fuel load, was not specifically included among the items reviewed by the SQRT. The applicant has, however, provided justification for interim operation (JIO) in his letters of August 1, 1984 and September 6, 1984, which, in the opinion of the staff, is adequate for operation until the first refueling outage. The basis of the staff conclusion is discussed as follows.

The RHRSW RPM detects high radiation levels in the cooling water effluent (RHRSW) from the RHR heat exchangers, in case of a heat exchanger tube leak of radioactive reactor coolant or suppression pool water to the RHRSW system. An RHRSW PRM high radiation signal actuates an alarm and automatically closes to RHR SW isolation valves and, if sensed at the loop discharge header, shuts off the RHRSW pump.

Seismic and dynamic qualification is completed for all of the equipment which is included in the PRM except for the log count rate meter (LCRM) located in the auxiliary control room for which the qualification test records are not readily available.

The justification for operating Unit 1 until the first refueling outage, with the qualification records of the LCRM incomplete, is as follows:

(1) Both RHR heat exchangers are seismically qualified, therefore, it is unlikely that a safe shutdown earthquake would result in a heat exchanger tube failure particularly in the early life of the tube materials. Consequently, it is acceptable for the first refueling cycle to consider that the RHRSW PRMs would not be required to perform the safety-related function during or following an SSE.

(2) There is a potential source of radiation leakage into the emergency service water (ESW) system, through the RHR pump seal coolers. The ESW system eventually merges with RHRSW system at the common return line. A similar PRM is located downstream of the return line to detect leakage. However, since both the ESW heat exchangers and the RHR pump seal cooler are seismically qualified, seismically induced leakage is unlikely as noted in (1) above.

(3) If the shutdown and isolation of one, or both, RHRSW supply systems results from false high radiation level PRM trip signals, the operator can manually bypass the signals and reopen the RHRSW isolation valves and restart the RHRSW supply pump(s).

(4) There is strong evidence that the LCRM was qualified by test, since bracing was added to the Limerick model LCRM.

For the reasons outlined above, the staff believes that the probability of a system failure associated with the LCRM is low enough to justify the safe interim operation of Limerick 1, up to the first refueling outage while the search for previous test records continues or an acceptable resolution is implemented. The JIO is therefore found to be acceptable.

Confirmation For New Load Definition

As a confirmation of the qualification loading input, the applicant was requested to verify that the staff-approved final new load definition has already been incorporated into his seismic and dynamic qualification program of safety-related equipment. In the letter of September 10, 1984, the applicant confirmed that FSAR (Revision 33) Sections 3.9.2.2 and 3.10 clearly indicate that the final load definition has been incorporated in the program for mechanical and electrical equipment, respectively, both in the NSSS and non-NSSS scopes. The staff finds this response to be acceptable.

Summary

Based on the SQRT site audit and the submittals from the applicant, the staff concludes that the applicant's equipment seismic and dynamic qualification program has been satisfactorily defined and implemented according to the intent of the current staff licensing criteria. The applicant must confirm in writing when the following actions are completed.

(1) Install the threaded taper pins for the RCIC and HPCI turbine assemblies prior to exceeding 5% power operation.

(2) Completely qualify, including full documentation, the RHR service water process radiation monitor prior to the end of the first refueling outage.

3.10.2 Operability Qualification of Pumps and Valves

To assure that the applicant has provided an adequate program for qualifying safety-related pumps to operate under normal and accident conditions the Equipment Qualification Branch (EQB) performs a two-step review. The first step is a review of Section 3.9.3.2 of the FSAR for the description of the applicant's pump and valve operability assurance program. This information is compared to Section 3.10 of the Standard Review Plan. The information provided in the FSAR, however, is general in nature and not sufficient by itself to provide confidence in the adequacy of the license's overall program for pump and valve operability qualification. To provide this confidence, the Pump and Valve Operability Review Team (PVORT), in addition to reviewing the FSAR, conducts an onsite audit of a small representative sample of safety-related pumps and valves and supporting documentation.

The onsite audit includes a plant inspection of the as-built configuration and installation of the equipment, a discussion of normal and accident conditions under which the equipment and systems must operate, and review of the qualification documentation (status reports, test reports, etc.). The two-step review is performed to determine the extent to which the qualification of equipment, as installed, meets the current licensing criteria as described in the Standard Review Plan 3.10. Conformance with these criteria provides an acceptable way of meeting the applicable portions of General Design Criteria 1, 2, 4, 14, and 30 of Appendix A to 10 CFR Part 50, as well as Appendix B to 10 CFR Part 50.

The onsite audit for the Limerick Generating Station Unit 1 was performed January 17-20, 1984. A walkdown was conducted to observe the as-built configuration of the selected equipment and to check for areas of deficient qualification. Whenever possible, the plant engineers described the features and operating procedures unique to the equipment. A representative sample of four pumps and six valves was chosen for the review. The sample included both NSSS and BOP equipment. The qualification documents were examined at the plant site, where the applicant maintains his central files.

During the PVORT review a number of concerns were raised. All of the specific concerns were satisfactorily resolved by the applicant during the audit by either supplying additional information or by demonstrating that the appropriate commitments are already addressed by administrative controls. The staff identified one generic concern as a result of the audit. The applicant had difficulty demonstrating the qualification of those components whose design parameters could be exceeded by postulated accident conditions. Shortly after the audit the applicant provided additional evidence that all active safety-related pumps and valves had been evaluated for their maximum service conditions. This submittal has been added to the docket file for Limerick and has adequately resolved all questions posed by the staff. Other generic topics discussed are of a positive nature.

Generic Findings

In preparation for the PVORT audit, the staff reviewed the Limerick FSAR Section 3.9.3.2 and the master list of seismic Category I equipment. The applicant provided sufficient information in these documents to allow the staff to conduct the onsite audit. Discussion with plant personnel during the audit further enhanced the staff's understanding of the equipment's functions and qualifications and qualification programs.

(1) There remained a small percentage of components whose qualification programs at the time of the PVORT audit were not complete or approved by the applicant. The staff required that the applicant submit for staff approval an updated master list or other form of confirmation which would verify that all safety-related equipment is qualified prior to fuel loading at Limerick. The applicant's submittal dated September 6, 1984 provided confirmation that all safety-related equipment has been qualified. The staff considers this generic concern adequately resolved.

(2) One generic operability concern resulted from the onsite evaluation of the Limerick qualification program for pump and valve operability. During the PVORT audit it was noted that the original design parameters for three of the ten selected components are less than the postulated peak pressure transient due to anticipated transient without scram (ATWS) scenario. The applicant did not have adequate documentation that the three components were capable of performing their function under the high pressure spike loads. Furthermore, the staff suspected that there were other equipment which may apparently exceed their design parameters. These concerns were presented to the applicant as a generic issue at the audit. The staff required the applicant to review all active safety-related pumps and valves to identify those equipment items for which the original design parameters were exceeded by the current accident or normal values. In each case for which the original design parameters were exceeded, the staff required the applicant to provide justification that pump and valve operability was not adversely affected.

    The applicant's supplementary qualification submittal dated April 5, 1984 adequately resolved all questions posed by the staff. Pressure integrity was generally addressed by designing equipment to the ANSI Standard pressure ratings which envelope the maximum service conditions. In addition, the applicant cited the ASME and ANSI code provisions for overpressure protection that allow 10 percent overpressure for events occuring less than one percent of the operating time.

    The applicant provided justification for excluding the peak ATWS conditions from the list of maximum service conditions. Hydrostatic tests and analyses were performed to assure pressure integrity of the primary pressure components during the initial peak ATWS condition. However, the GE design basis for ATWS indicates that none of these components are required to perform any active safety-related functions for this condition. Furthermore, the subsequent long-term peak ATWS conditions, which are considered for equipment operability, are less severe than the component design conditions. Consequently, the peak ATWS transient conditions do not adversely affect operability and are not included in the maximum service conditions listed for active safety-related pumps and valves. All components exposed to ATWS transient conditions have been evaluated as acceptable in accordance with GE design basis for ATWS.

The applicant has clearly stated in the submittal that all BOP and NSSS active safety-related pumps and valves have been evaluated against the worst case normal or accident conditions and have been declared acceptable. This supplementary qualification submittal has been added to the NRC licensing file for Limerick. The staff's generic concern about equipment operability has been adequately resolved.

(3) A major concern was the number of quality hold and temporary modification tags that were attached to many pieces of installed equipment. The Limerick systems engineers explained that the plant was then undergoing construction and some preoperational flush tests. Test procedures were described which specified test sequences, system line-up procedures, and temporary equipment changes. The utility staff described a program of tracking the plant operation state. The execution of this program satisfactorily addressed the concern of the operational status of plant equipment. Documentation of changes showed full accountability of the equipment status and resolved the concerns raised by the staff.

(4) The applicant presented a brief orientation lecture on maintenance and surveillance. In addition, PVORT made a limited review of the corresponding documentation. Limerick has a computer-based maintenance program, which appears to be very comprehensive and which includes many excellent features. Some of these include: (a) incorporating all of the pertinent data provided by the vendors, such as aging information, spare parts, and maintenance schedules, (b) continuous monitoring of pumps over 50 horsepower in order to detect and analyze trends which may be indicative of degradation and to implement a vibration analysis program, (c) providing a closed-loop check by the quality assurance group to inspect and verify maintenance and other related activities performed on the equipment, and (d) analyzing equipment upon removal to help in determining changes in the inspection and replacement schedules. Furthermore, Limerick voluntarily participates in the Nuclear Plant Reliability Data System (NPRDS). The staff concludes that the qualification considerations have been incorporated into the maintenance, surveillance, preoperational testing, and inservice test programs and finds the applicant's methods acceptable.

(5) The startup and preoperational test programs appear to be very comprehensive and will soon be implemented. Over 80% of the construction tests have been completed, approximately 40% of the equipment has been turned over to the applicant and some preoperational tests have been completed or were in progress at the time of the audit. The initial test program consists of generic procedures for system lineup and component calibration, and specific procedures for equipment functional performance. A detailed program of documentation and administrative procedures addresses temporary

alterations of equipment in preparation for testing, equipment status, consideration for retest, and test compliance. The staff concludes that the procedures, as they relate to equipment status and qualification, use the quality assurance criteria of Appendix B to 10 CFR Part 50 and are acceptable.

Specific Concerns

A number of minor concerns, noted during the Limerick walkdown, were satisfactorily resolved during the audit. Many of these issues were attributed to the construction and preoperational test schedules in progress during the review. Other concerns were satisfactorily addressed by administrative controls already in effect. The applicant was able to explain and justify the presence of quality control tags, temporary equipment modifications, test preparations, and followup procedures. PVORT made a check of the applicant's documentation system by requesting at short notice and reviewing in detail the appropriate quality tags, startup work requests, test reports, and related document controls. The following examples illustrate the manner by which the applicant satisfactorily addressed specific concerns at the Limerick plant.

(1) The HPCI check valve (M55-1F005) was observed to have the internals removed and a temporary bonnet installed in order to conduct a system flush. The removal and temporary modification tags and quality control prcedures appear to be adequate to address this temporary configuration. Although the equipment as observed was not operable, the applicant appears to have adequate procedures already in place which will address the operability issue prior to fuel load once the preoperational tests are completed.

(2) The reactor recirculation sample globe valve (HV-43-1F019) was observed to have the main air supply line to the regulator disconnected and not tagged. The disconnected tubing was taped to the valve actuator box. The plant engineer explained that the line had been disconnected briefly to permit the air lines to be blown clear before commencing tests on other equipment. The engineer indicated that the air supply lines would be checked prior to testing. Although the equipment as observed was not operable, the applicant appears to have adequate procedures in place to address the operability issue.

(3) The motor-operated butterfly valve (HV-11-011) was observed to have a defective weld in the drain line. The quality control hold tag and administrative controls appeared to be adequate. The defective weld was corrected before completion of the PVORT audit. The valve, as observed, was operable.

(4) The standby liquid control pump and motor (1AP 208) was observed to be decoupled. The applicant stated that the vendor had not yet completed his construction test activities and that the pump was decoupled for alignment checks. After completion of the construction activities and appropriate documentation, the equipment will be turned over to the applicant's startup group for approval prior to the preoperational test. While the pump, as observed, was not operable, it is concluded that the applicant has administrative controls already in place which will satisfactorily address the operability issue.

The staff concludes that the applicant's procedures for tracking equipment status and the immediate attention to identifying safety-related equipment are conducted in an orderly and disciplined manner and are acceptable.

Conclusion

The Equipment Qualification personnel for Limerick are dealing with the equipment qualification issue in a very positive manner. The staff has reached this conclusion because the applicant has: (a) provided adequate documentation to demonstrate qualification of safety-related pumps and valves, (b) established administrative programs to determine, monitor, and maintain equipment operability for the lifetime of the plant, (c) demonstrated an adequate central file system by the timely retrieval of information requested by the staff during the audit, (d) corresponded closely with equipment suppliers to discuss and evaluate details of construction, test procedures, and plant operation, and (e) demonstrated overall accountability by committing their appropriate personnel to implement these programs.

Based on the results of the site review performed at Limerick January 17-20, 1984 and the subsequent submittals by the applicant to resolve issues identified from the site review, we concluded that an appropriate pump and valve operability qualification program has been defined. The continuous implementation of this overall program should provide adequate assurance that the safety-related functions will be performed as needed.

4 REACTOR

4.2 Fuel System Design

4.2.1 Design Bases

4.2.1.1(6) Fuel and Poison Rod Pressures

In Section 1.10 of the SER the staff provided a list of issues for which a license condition may have been desirable to ensure that staff requirements are met during plant operation unless satisfactory resolution was otherwise reached prior to issuance of the license. One such issue was on internal fuel rod pressures.

In the SER the staff stated that Limerick fuel did not meet the SRP criterion that the internal fuel rod pressure be less than or equal to the coolant system pressure for all burnups considered as required by SRP Section 4.2.II.A.1.F.

In a letter from J. S. Charnley (GE) to C. O. Thomas (NRC), "Response to Request for Additional Information on Proposed Amendment to GE Licensing Topical Report, NEDE-24011-P-A," dated December 19, 1983, GE stated that the criterion proposed by General Electric which relates cladding creepout rate to fuel swelling rate will not (a) result in fuel system damage during normal operation and AOO's, (b) prevent control rod insertion, (c) lead to loss of coolable geometry, or (d) result in an underestimate of the number of fuel failures in or radiological consequences of postulated accidents.

In this GE submittals, GE describes a design basis for rod pressure in which the effects of fuel rod internal pressure during normal steady-state operation will not result in fuel failure due to excessive cladding pressure loading. GE contends that a rod internal pressure limit of less than or equal to the RCS pressure is not necessary. Instead, GE proposes that the rod pressure be limited so that the instantaneous cladding creepout rate due to internal pressure greater than RCS pressure is not expected to exceed the instantaneous fuel swelling rate.

To demonstrate that this proposed criterion is acceptable in terms of items (a) through (d) above, GE demonstrates that for the design basis transients and accidents of interest in a BWR, either the cladding does not heat up significantly or the existing fuel damage criteria used are still applicable when the initial fuel rod internal pressure exceeds the initial RCS pressure.

In the case where the cladding does not heat up significantly, that is, the safety limit MCPR is not exceeded, there is no significant change in the fuel rod geometry so that control rod insertion and bundle coolability will be maintained.

For those events in which the cladding does heat up signficantly above its normal temperature, GE has demonstrated that there are other criteria which assure that conditions (a) through (d) above will not occur. For example, the LOCA event is governed by the criteria set forth in 10 CFR 50.46 that the cladding temperature will not exceed 2200°F, the maximum amount of local oxidation on any fuel rod will not exceed 17% and that a coolable geometry will be maintained. These criteria are independent of the initial internal pressure of the fuel rod. However, the internal pressure of the fuel rod is taken into account explicitly in determining the stored energy and in calculating the amount of fuel rod swelling and rupturing. In addition, the number of failed fuel rods assumed for radiological calculations is 100% of those in the core. Therefore, a rod internal pressure greater than the RCS pressure will not result in underestimating the radiological consequences of a LOCA. Therefore, a fuel rod internal pressure greater than RCS pressure is acceptable for LOCA.

Similarly GE has evaluated the rod drop accident and has demonstrated, in response to a staff question, that the criterion for fuel failure in a rod drop accident is still applicable as stated in a letter from J. S. Charnley (GE) to R. Lobel (NRC), "NRC Questions on Amendment 7 to NEDE-24011-P-A," Dated April 2, 1984.

The staff therefore finds the GE criterion for fuel rod internal pressure to be acceptable.

In a letter dated August 17, 1984 the applicant stated that the GE submittal was applicable to the Limerick initial core. Therefore, the staff concludes that this issue is resolved and does not need to be addressed by a license condition for Limerick.

4.2.3 Design Evaluation

4.2.3.2(4) Overheating of Fuel Pellets

In the SER the staff stated that it had requested the applicant to confirm the adequacy of applicable information on this subject to the Limerick plant. The staff noted that fuel melting temperature as a function of exposure (burnup) and gadolinia content (of burnable poison rods) is discussed in Section 2.4.2.5 of NEDE-24011. In that report, General Electric stated that fuel melting is not expected to occur during normal operation, and that prediction is based on fuel temperature calculations performed with a model described in the proprietary supplement to Amendment 14 of GESSAR (STN 50-447). While limited melting during certain events such as an uncontrolled control rod withdrawal is permissible, such melting is not predicted to occur.

The staff has reviewed the UO_2 properties (thermal conductivity and melting point) that are important in reaching this conclusion and agree that UO_2 melting will not be a problem at Limerick during normal operation and anticipated transients as long as the 1 percent plastic strain criterion discussed in SRP Section 15.4.2 is not exceeded. In the SER the staff also noted that the effects of gadolinia concentration on thermal conductivity and melting temperature were addressed in an unreviewed GE topical report on gadolinia fuel properties (NEDE-20943). That report has been replaced by another topical report (NEDE-23785-1), which described revised fuel thermal performance methods and (Appendix B) gadolinia properties. The more recent report has been reviewed and approved by the NRC staff.

General Electric has stated (J. S. Charnley to L. S. Rubenstein dated February 2, 1984) that gadolinia properties described in Appendix B of NEDE-23785-1 are generically applicable to new plants such as Limerick and has also confirmed that the applicable limits for overheating of gadolinia fuel remain valid. Because these limits were previously found acceptable to Limerick and because the applicant has utilized approved methods (and gadolinia properties) to show that these limits continue to be met, this issue is considered resolved.

4.4 Thermal-Hydraulic Design

4.4.4 Thermal Hydraulic Stability

Stability Test Data

The staff recently became aware of new stability test data which demonstrated the occurrence of limit cycle neutron flux oscillations at natural circulation and several percent above the rated rod line.

The oscillations were observable on the average power range monitors (APRM's) and were suppressed with control rod insertion. It was predicted that limit cycle oscillations would occur at the operating condition tested; however, the characteristics of the observed oscillations were different than those previously observed during other stability tests. Namely, the test data showed that some LPRM indications oscillated out of phase with the APRM signal and at an amplitude as great as six times the core average.

The General Electric Company has prepared SIL #380 for release to alert utilities of these new data and to recommend actions to avoid and control abnormal neutron flux oscillation. The applicant for the Limerick plant proposed technical specifications to be consistent with GE's recommendations in SIL #380 to protect against the potential for thermal hydraulic instability in accordance with GDC 12. The principal changes to the technical specifications are the following.

1. When operating with one or no recirculation pumps in operation, the plant will immediately initiate action to reduce thermal power to less than or equal to a specific limit.

2. When in two-loop operation at total core flow rates less than 45% of rated core flow and at thermal power greater than a specific limit, and with the APRM or LPRM neutron flux noise levels greater than three times their established baseline levels, restore the noise level to within the required limits within 2 hours. This may be done either by increasing core flow to greater than 45% of rated core flow or by reducing thermal power to less than or equal to the specific limit.

The staff has reviewed these proposed technical specifications and has found that they are consistent with the recommendations in SIL-380 and acceptably resolve the thermal-hydraulic stability concern for Limerick Units 1 and 2 assuming long-term single-loop operation is not permitted. Should such operation be requested in the future, the staff will evaluate the Limerick Units 1 and 2 Technical Specifications to determine if additional modifications are required.

Thermal-Hydraulic Analysis Methods

In order to assure that the thermal-hydraulic safety design criteria regarding the MCPR limits and thermal-hydraulic stability margin will be met for operations beyond Cycle 1 core, the following license condition on Limerick Units 1 and 2 was specified as stated in Sections 4.4.4 and 4.4.9 of the SER.

> Operating beyond Cycle 1 is not permitted until results of a stability analysis and calculated MCPR are provided for the additional cycles of operation.

The applicant was informed by the SER that the existing analyses do not support operation beyond Cycle 1 and has, by letter dated October 4, 1984, agreed to submit for staff review the similar analytical results including the MCPR limits and thermal-hydraulic stability margin, as part of the reload licensing application beyond Cycle 1 core operation. Based on the agreement, the staff has concluded that the above license condition is not necessary. The staff will review the analytical results when they become available, and provide the evaluation results as appropriate to support operation beyond Cycle 1.

4.6 Functional Design of Reactivity Control Systems

In Section 1.10 of the SER the staff provided a list of issues for which a license condition may have been desirable to ensure that staff requirements are met during plant operation. One such issue was on BWR scram system piping. The SER noted that NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping" had been issued in August 1981 and that the BWR Owners Group had submitted generic responses concerning this issue.

Since this is a multiplant action item, the staff has not made a determination as to what design changes, if any, are necessary for Limerick Unit 1 until the review of the BWR Owners Group responses is complete. In a letter dated June 28, 1983 the applicant committed to implement all actions and modifications agreed to between the staff and the BWR Owners Group on this issue by the first refueling outage scheduled 12 months after the agreement has been established. In response to further discussions with the staff on this issue, the applicant also provided commitments by letter dated October 15, 1984, relating to the BWROG's letter of May 10, 1984. The staff finds the applicant's commitment acceptable pending the completion of the resolution of this multi-plant action item.

5 REACTOR COOLANT SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

5.2.4.3 Evaluation of Compliance with 10 CFR 50.55a(g) for Limerick Unit 1

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

This evaluation supplements conclusions in this section of the SER, which addresses the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). Review has been completed of the information presented in the FSAR through Revision 34 dated July 1984, the Preservice Inspection (PSI) Program submitted September 24, 1982 and June 30, 1983, the information obtained at a public meeting at Bethesda, MD on August 31, 1983 and, as a result of this meeting, the Applicant's letters dated October 5, 1983 and December 28, 1983. In submittals dated July 17, 1984, August 7, 1984, August 23, 1984, and August 28, 1984 and August 30, 1984, the Applicant requested relief from ASME Section XI Code requirements which have been determined to be not practical to perform. These relief requests were supported by information pursuant to 10 CFR 50.55(a)(2)(i). Therefore, the staff evaluation consisted of reviewing these submittals and determining if relief from the Code requirements were justified. Pursuant to 10 CFR 50, Paragraph 50.55a(a)(2), the staff has allowed relief from the impractical requirements that, if implemented, would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. The detailed evaluation of relief requests is included in Appendix N to this report.

The reactor pressure vessel (RPV) was examined in accordance with the 1980 Edition of the ASME Code, Section XI, including Winter 1980 Addenda and Regulatory Guide 1.150, Revision 1. An enhanced examination has been completed on the circumferential welds and the full length of all intersecting longitudinal welds in the beltline region. Data was evaluated at 20% DAC. Calibration for the examination of the RPV inside surface utilized a calibration block with both notches and holes at the clad to base metal interface. The Applicant states that a 1/8 inch diameter side drilled hole at the clad to base metal interface is readily detectable. The examination was performed using a mechanized ultrasonic search-unit assembly. The position of the search-unit assembly is tracked by 18 acoustic emission (AE) transducers precisely located at the RPV. The RPV is divided into zones which are monitored by 3 AE transducers in each zone. An odometer on the search-unit assembly and a closed circuit television system provide backup information for tracking. The Applicant indicated that the only equipment modifications required to meet Regulatory Guide 1.150 were in the processing and data acquisition systems and not in the mechanical portions of the equipment. As a result of the August 31, 1983 meeting, the Applicant submitted summary reports dated December 28, 1983 and July 17, 1984 describing the method of compliance with Regulatory Guide 1.150 including the method of procedure qualification. The staff concludes the Applicant meets the intent of Regulatory Guide 1.150 for boiling water reactors (1) by qualifying the examination procedures to assure finding service-induced flaws on the inside surface of the vessel, and (2) by documenting all areas on the RPV where the PSI requirements, defined in Section XI of the ASME Code, that have been determined to be impractical and providing a supporting technical justification. The RPV examination results, including plant-specific areas where the Code requirements cannot be met along with a supporting technical justification, were included in the Applicant's July 17, 1984 submittal. Based on the above review, the staff considers the Limerick Unit 1 reactor pressure vessel examination in compliance with ASME Code Section XI, meets the intent of Regulatory Guide 1.150, Revision 1, and therefore, is in compliance with 10 CFR Part 50, Paragraph 50.55a(g).

In a letter dated November 4, 1981 and in a second letter dated October 5, 1983, which resulted from the August 31, 1983 meeting, the Applicant discussed the reactor coolant pressure boundary (RCPB) piping and fitting material as related to conformance with the material selection and process guidelines set forth in NUREG-0313, Revision 1, dated July 1980. The results of the review for conformance with NUREG-0313 was presented in Section 5.2.3 of NUREG-0991. All welds in the nonconforming portions of Class 1 systems, including flued head to valve welds, are examined using both ultrasonic and liquid penetrant techniques as required by the ASME Code Section XI.

Based on review of the Applicant's submittals, the staff has determined that the Limerick Generating Station Unit 1 PSI Program is acceptable and that the review is considered to be completed.

The initial Inservice Inspection (ISI) program has not been submitted by the Applicant. The program will be evaluated after the applicable ASME Code Edition and Addenda can be determined based on Paragraph 50.55a(b) of 10 CFR Part 50, but before the first refueling outage when inservice inspection commences.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

Compliance with Appendix G, 10 CFR 50 For Unit 1

In the SER the staff determined that the requirements of the then current Appendix G had been met except for specific matters discussed in the SER. Since publication of the SER in August 1983 Appendix G has been revised. The revision became effective on July 26, 1983.

In lieu of the requirements in Appendix G which were discussed in the SER, the revised Appendix G requires that the fracture toughness program meet the ASME Code edition and addenda, as permitted by Paragraph 50.55a, 10 CFR 50. As discussed in the SER, the fracture toughness test program for LGS-1 does not comply with the ASME Code fracture toughness requirements, required by Paragraph 50.55a. However, Appendix G permits, for a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition, that the fracture toughness data and data analyses may be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of the Appendix. LGS-1 was constructed to an ASME Code which was earlier than the Summer 1972 Addenda of the 1971 Edition.

In the SER the staff presented an evaluation of the fracture toughness data and data analyses, which was presented by the applicant. The staff considers that the data presented by the applicant demonstrates that the fracture toughness properties of the ferritic reactor coolant pressure boundary materials are equivalent to that required by the Appendix. Hence, exemptions to Appendix G are not required. A letter dated September 25, 1984 from the Director, Office of Nuclear Reactor Regulation to the applicant finds acceptable the applicant's method of demonstrating equivalency which was documented in the FSAR through Amendment No. 20 and in a letter from the applicant dated June 14, 1983.

Compliance With Appendix H, 10 CFR 50 For Unit 1

In the SER the staff determined that the requirements of the then current Appendix H had been met except for specific matters discussed in the SER. Since publication of the SER in August 1983 Appendix H has been revised. The revision became effective on July 26, 1983.

In the SER the staff indicated that the applicants CVN impact surveillance plate material did not conform with the specimen orientation and limiting materials requirements of the 1973 edition of ASTM E 185. The now-revised Appendix H requires the surveillance program to comply with the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. Since the Limerick reactor vessel was procured to an earlier edition of the ASME Code than 1973, the revised Appendix H permits the surveillance program to comply with an earlier edition of ASTM E 185 than the 1973 edition. The Limerick Unit 1 surveillance plate material complies with the specimen orientation and limiting requirements of the earlier edition of ASTM E 185. Hence, the staff concludes that Limerick Unit 1 complies with the revised Appendix H requirements.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.3 Secondary Containment Functional Design

Revised SGTS Drawdown Analysis

In Section 6.2.3 of the SER the staff indicated that the applicant had performed a post-LOCA pressure transient analysis for the reactor enclosure building and determined the length of time the pressure would exceed -0.25 inch w.g. This drawdown time was based on drawing down both units' reactor enclosure buildings at the same time. The staff also stated, in the SER, that the applicant's analysis had been reviewed against the criteria in SRP 6.2.3 and found acceptable.

In a letter dated August 2, 1984, the applicant provided a revised analysis which reflects an increase of the inleakage rate for the reactor enclosure design from 50 to 100 percent free volume per day and only drawing down the Unit 1 reactor enclosure, which will be the case prior to Unit 2 operation.

Based on the above assumptions, and assuming a standby gas treatment system (SGTS) maximum flow rate of 2800 cfm, the applicant's drawdown analysis indicates that the secondary containment (reactor enclosure) pressure would exceed -0.25 inch w.g. for 2.25 minutes from the time the SGTS receives its initiation signal. The maximum SGTS flow rate of 2800 cfm corresponds to the first three minutes after an accident, when the reactor enclosure recirculation system (RERS) is not yet in operation.

The applicant also stated that prior to fuel load for Unit 2, certain design modifications will be made to ensure a 2.25-minute drawdown for the case of two-unit operation.

The applicant's analysis of the post-LOCA pressure transient in the reactor enclosure indicates that the heat load resulting from RERS initiation at 3 minutes will not cause the pressure in the enclosure building to return to a positive pressure. To verify this analytical result, the applicant will perform a one-time surveillance test: (a) by operating the standby gas treatment system for one hour and verifying that it will drawdown the secondary containment to a vacuum of greater than or equal to 0.25 inch of water gauge in less than or equal to 135 seconds; and (b) by initiating the reactor enclosure recirculation system at 3 minutes and maintaining a vacuum of greater than or equal to 0.25 inch of water gauge in the secondary containment during the operation of the RERS.

The staff has reviewed the applicant's analysis and proposed test against the criteria in SRP 6.2.3 and finds it acceptable.

SGTS Connection to Refueling Floor

In a September 21, 1984 letter, the applicant requested a schedular exemption from the requirement of 10 CFR 50, Appendix A, GDC 61 as it relates to the filtering of radioactive gases in the refueling floor zone under postulated accident conditions.

To preclude the release of radioactivity to the refueling floor zone, the applicant in a July 26, 1984 letter made the following commitments until such time as the refueling floor is connected to the SGTS:

a. Operations involving removal of the primary containment and RPV heads after initial criticality are prohibited without specific, prior NRC approval.

b. The handling of any loads (other than the RPV head, dryer or separator) over irradiated fuel will be carried out in accordance with the single failure criteria of NUREG-0612.

c. Operations involving handling and storage of irradiated fuel will not be undertaken.

Prior to the applicant's exemption request, the staff in Section 6.2.3 of SER Supplement No. 2 evaluated the effects of the refueling floor zone not being connected to the SGTS prior to the movement of irradiated fuel and determined this to be acceptable provided the connection of the refueling floor zone to the SGTS was made a licensing condition.

Based on the above discussion and the applicant's commitments, we find that a schedular exemption from the requirement of GDC 61 is justified since it does not cause any undue risk to the health and safety of the public. The licensing condition related to this issue as stated in SER Supplement No. 2 will be imposed.

6.2.4 Containment Isolation System

6.2.4.2 General Design Criterion 56

A. Containment Isolation Provisions for Hydrogen Recombiner System

The applicant in a September 21, 1984 letter requested a schedular exemption from the requirements of GDC 56 as they relate to the containment isolation provisions for the hydrogen recombiner system.

In Section 6.2.4 of SER Supplement No. 1 the staff evaluated the applicant's rationale for not having two automatic isolation valves in each of the hydrogen recombiner lines penetrating the containment and concluded that the applicant had provided adequate justification for operation for Limerick Unit 1 through the first cycle. The applicant in a September 22, 1983 letter committed to install additional isolation valves before startup after the first refueling outage. The staff in Supplement No. 1 identified this as a deviation from GDC 56 which required the granting of an exemption.

Based on the applicant's September 21, 1984 letter requesting an exemption from GDC 56 and our evaluation contained in SER Supplement No. 1, the staff has determined that a schedular exemption from the requirement of GDC 56 is justified since it does not cause any undue risk to the health and safety of the public.

B. Drywell Chilled Water (DCW) and Reactor Enclosure Cooling Water (RECW) Systems

The applicant in a September 21, 1984 letter also requested a schedular exemption to the requirements of GDC 56 as they relate to the isolation provisions for the DCW and RECW systems.

In Section 6.2.4.2 of the SER the staff evaluated the applicant's basis for plant operation until the first refueling outage without having automatic closure by diverse containment isolation signals for DCW outboard containment isolation valves and the RECW containment isolation valves (Supply and Return). The staff determined that since: (1) these lines do not open directly to the containment atmosphere or to the reactor coolant boundary; (2) these lines are designed to withstand a seismic event; and (3) the applicant has committed to provide special interim operating instructions to isolate these lines should a LOCA occur, operation of the plant during the first cycle without automatic isolation of these valves is acceptable.

Based on the applicant's September 21, 1984 letter requesting an exemption and our evaluation contained in the SER, the staff has determined that a schedular exemption from the requirement of GDC 56 during the first cycle of operation is acceptable since it does not cause any undue risk to the health and safety of the public.

6.2.6 Containment Leakage Testing Program

In the SER the staff concluded that the applicant's proposed leak testing program meets the requirements of Appendix J to 10 CFR 50 and is, therefore, acceptable or that the applicant has provided acceptable justification for the deviation from the explicit requirements of Appendix J. The following identifies each proposed deviation and provides the staff's evaluation of the proposed deviation.

1. <u>Main Steam Isolation Valves (MSIVs)</u>

Excluding the MSIV leakage from the summation for local rate tests, which is different from the explicit requirement of 10 CFR 50, was discussed and found acceptable in Section 6.2.6.1 of the SER. In that SER the staff determined that an exemption from the requirements of Appendix J was justified for this deviation.

2. <u>Traversing Incore Probe (TIP) System</u>

An exemption from the requirement to test the TIP shear valves in accordance with Appendix J, Paragraphs III.H.1 and III.B.2, and the proposed alternative testing of these valves was discussed and found acceptable in Section 6.2.6.2 of the SER.

3. RHR Relief Valve Discharge

A one-time exemption from the requirement to perform local leak rate testing on seven RHR relief valves was discussed and found acceptable in Section 6.2.6.5 of the SER.

4. Air Lock Testing

Appendix J, Paragraph III.D.2.(b)(ii) requires that "Air locks opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods at not less than Pa."

In lieu of this requirement, the applicant requested that the overall air lock leakage test at Pa be conducted only when maintenance has been performed on the air lock that could affect the air lock sealing capability. The applicant stated that a full pressure test at Pa will require installing strongbacks on the inner door which is a cumbersome process requiring at least 12 hours. The applicant further stated that the air lock leak tightness is assured, if no maintenance which could affect the ability of the air lock to seal has been performed, by compliance with the six month periodic test requirements of paragraph III.D.2(b)(i) and the three day test requirements of paragraph III.D.2(b)(iii) of Appendix J. The staff agrees with the applicant's rationale; however, the staff has proposed and the applicant has agreed to verify seal leakage to be within the Technical Specification limit prior to establishing primary containment integrity when the air lock has been used and no maintenance has been performed on the air lock. This will be done by pressuring the gap between the door seals to 10 psig. The staff finds that an exemption from the requirement of paragraph III.D.2(b)(ii) of Appendix J is warranted since no increase in air lock leakage is to be expected as a result of that exemption.

6.2.7 Fracture Prevention of Containment Pressure Boundary

Our safety evaluation review assessed the ferritic materials in the Limerick Generating Station Units 1 & 2 containment system that constitute the containment pressure boundary to determine if the material fracture toughness is in compliance with the requirements of General Design Criterion 51, "Fracture Prevention of Containment Pressure Boundary."

GDC 51 requires that under operating, maintenance, testing and postulated accident conditions, (1) the ferritic materials of the containment pressure boundary behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The Limerick Generating Station containment system is a reinforced concrete structure with a thin steel liner on the inside surface which serves as a leak-tight membrane. The ferritic materials of the containment pressure boundary which were considered in our assessment are those which have been applied in the fabrication of the drywell head, equipment hatch, personnel locks, penetrations and fluid system components, including the valves required to isolate the system. These components are the part of the containment system which are not backed by concrete and must sustain loads during the performance of the containment function under the conditions cited by GDC 51.

We have determined that the fracture toughness requirements contained in ASME Code editions and addenda typical of those used in the design of the Limerick Generating Station containment may not ensure full compliance with GDC 51 for all areas of the containment pressure boundary. As a result, we have elected to apply in our licensing reviews of ferritic containment pressure boundary materials, the criteria for Class 2 components identified in the Summer 1977 Addenda of Section III of the ASME Code. Because the fracture toughness criteria that have been applied in construction typically differ in Code classification and Code edition and addenda, we have chosen the criteria in the Summer 1977 Addenda of Section III of the Code to provide a uniform review consistent with the safety function of the containment pressure boundary materials. Therefore, we reviewed the materials of the components of the Limerick Generating Station containment pressure boundary according to the fracture toughness requirements of the Summer 1977 Addenda of Section III for Class 2 components.

Considered in our review were components of the containment system which are load bearing and provide a pressure boundary in the performance of the containment function under operating, maintenance, testing and postulated accident conditions as addressed in GDC 51. These components are the drywell head, equipment hatch, personnel airlocks, penetrations and elements of specific containment penetrating system.

Our assessment of the fracture toughness of materials of the Limerick Station containment pressure boundary was based on the metallurgical characterization of these materials and fracture toughness data presented in NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," USNRC, October 1979, for comment, and ASME Code Section III, Summer 1977 Addenda, Subsection NC.

The metallurgical characterization of these materials, with respect to their fracture toughness, was developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG-0577 above and the Summer 1977 Addenda of the ASME Code Section III, provided the technical basis for our evaluation of compliance with GDC 51.

Based on our review of the available fracture toughness data and materials fabrication histories, and the use of correlations between metallurgical characteristics and material fracture toughness, we conclude, with one condition, that the ferritic materials of the components of the Limerick Generating Station containment pressure boundary meet the fracture toughness requirements that are specified for Class 2 components by the Summer 1977 Addenda of Section III of the ASME Code. Compliance with these Code requirements provides reasonable assurance that materials of the Limerick Generating Station reactor containment pressure boundary will behave in a nonbrittle manner, that the probability of rapidly propagating fracture will be minimized and the requirements of GDC 51 are satisfied. The condition relates to the feedwater check valves (1F074 A&B and 2F074 A&B) which is addressed below.

Limerick 1 & 2 Feedwater Check Valves (1F074 A/B:2F074 A/B)

Our review identified 24" feedwater check valves 1F074 A&B and 2F074 A&B as parts of the reactor containment pressure boundary. The cast bodies of these valves are known to contain shrinkage flaws which have been known to propagate in service. Because of the presence of these flaws and the uncertainty related to their propagation in service, we were unable to conclude, relative to fracture toughness, that sufficient margin of safety existed under the limiting environmental condition to be experienced by these valves, viz., 1180 psi at 42°F postulated for HPCI, as identified by the applicant, when these valves are called upon to serve as a containment pressure boundary.

In accordance with our review practice, conformance with GDC 51 is assured when the lowest service temperature is 30°F above the NDT of the material. In this case, therefore, if we could be assured that the NDT was at or below 12°F, we could consider the component to be acceptable.

The applicant submitted in support of his position regarding the acceptability of these valves, Bechtel Tech Report No. 1183-05EV, Revision 2, dated May 1984, and titled "Acceptability of Class 1 24-inch Feedwater Check Valves." We have reviewed the report within the context of the compliance with GDC 51 requirements.

Although the applicant has submitted Charpy data that he feels supports an NDT of 10°F or below, we have reservations about the NDT-Charpy relationship for this material, and also have concerns that test on separately cast "keel blocks" may not conservatively represent the properties of the actual castings. Using NUREG-0577 recommendations and considering the guidance given in Table NC-2311(a)-1 of the Summer 1977 Addenda of the Boiler and Pressure Vessel Code, we have concluded that it is reasonable and conservative to assume that these valve body castings have an NDT of 30°F, and therefore do not meet our basic criterion.

The applicant performed a fracture mechanics calculation that concluded that wide margins against failure by brittle fracture exist. We have some reservations regarding the assumptions used to determine the critical K_{IC} of the material. The approach was to determine the K_{IC} by a correlation with Charpy energy values. The correlation method used has not received universal acceptance.

Because we were not satisfied with the applicant's fracture mechanics analysis, we performed our own independent calculations. Instead of the correlation with Charpy values, we chose to use the method recommended in Appendix G of Section III of the Code, supplemented by additional calculations using Section XI. In these methods, the fracture toughness of the material is given by K_{IR}-temperature curves, and K_{IC}-temperature curves, that are indexed to the RT_{NDT} of the material. The RT_{NDT} is basically the same as NDT for material with normal Charpy properties.

We used the stress levels furnished by the applicant for our analysis as these appeared to be somewhat more conservative than our own calculations indicated. We also chose to use the assumed flaw size selected by the applicant, one inch deep by 3.5 inches long. This size, in our opinion, bounds the dimensions of acceptable shrinkage (severity level 2) with margin for postulated growth during operation. The results of our calculations are as follows.

For the Section III Appendix G calculations, the K_{IR} (lower bound estimate of dynamic or arrest toughness) at RT_{NDT} + 12°F is 41 Ksi $\sqrt{in}$. The applied K_I, including both membrane and bending stress, is calculated to be 24.3 Ksi $\sqrt{in}$. This indicates a margin of 41 divided by 24.3, or a factor of 1.7 against failure under dynamic conditions at 1180 psi and 42°F. Appendix G requires a margin of a factor of two on pressure stress, but only a factor of one on bending stress (assuming the bending stress is not caused by pressure). In this case, it was stated by the applicant that some of the bending stress could be pressure-related, so we assume that a factor of two should be provided for the combined stresses. Applying a factor of two on the membrane stress of 10.4 psi and a factor of one on the bending stress of 10.2 psi, the total K_I is 38.3, which is less than the K_{IR} of 41, so the Code recommended margins would be met if bending stress were independent of pressure.

Although these calculations indicated that the valve body would come very close to meeting the Code Appendix G criteria, we recognized a possible non-conservatism in the approach. The Code states that the K_{IR} curve may be used for steel with minimum specified yield strengths of 50 Ksi and less. As the minimum yield strength for the valve body material is 35 Ksi, the use of the K_{IR} curve is acceptable to the Code. Nevertheless, the principle underlying the development of the K_{IR} curve and the temperature correlation with NDT implies that the K_{IR} values should be reduced by the ratio of the yield strength of the material of concern and that of the steels used to develop the correlation. Because all of the steels used to establish the K_{IR} curve had minimum yield strengths of 50 Ksi, we performed calculations in which the assumed K_{IR} at 12°F was reduced by the ratio of 35 divided by 50. In this case, the K_{IR} value is 28.7 Ksi $\sqrt{in}$, whereas the calculated K_I was 24.3 Ksi $\sqrt{in}$, providing a margin of only 1.2 instead of the factor of two specified in Appendix G.

We also performed a more realistic calculation using the lower bound K_{IC} values from Section XI of the Code. This assumes the more probable quasistatic loading condition. The K_{IC} at RT_{NDT} + 12°F is 69 Ksi $\sqrt{in}$, giving a margin of a factor of 2.8 against failure. Reducing the K_{IC} by the ratio of the yield strength (as discussed above) still results in a margin of a factor of 2.0.

We have concluded from these calculations that although the valve body may not quite meet the Appendix G requirements, even our most conservative approach still shows some safety margin. Adequate margin against failure (at least a factor of two) will exist under the most probable loading conditions.

These conclusions assume that the flaw size assumed will not be exceeded significantly. However, service experience on similar castings has disclosed that normal, acceptable shrinkage may be extended by cracking during service. We, therefore, recommended to the applicant that these valves be inspected for surface cracks on the inside and outside surfaces at the first refueling outage and at other times when the valve is disassembled for maintenance. The applicant has committed, by letter dated October 12, 1984, to including this augmented inspection by surface examination or other methods acceptable to the staff, which will be determined during the staff's review of the inservice inspection program.

We have concluded that the results of our evaluation and the augmented inservice inspection program for these valves will provide reasonable assurance of compliance with the requirements of GDC 51. It will be confirmed by the augmented ISI that the shrinkage flaws existing in the valve bodies on entering service have not propagated to either of the surfaces. Should the augmented ISI disclose that these flaws have propagated to either of the surfaces, the valves are then to be replaced by the licensee.

6.6 Inservice Inspection of Class 2 and 3 Components

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

6.6.3 Evaluation of Compliance with 10 CFR 50.55a(g) for Limerick Generating Station Unit 1

This evaluation supplements conclusions in this section of the SER, which addresses the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).

Review has been completed of the information presented in the FSAR through Revision 34 dated July 1984, the PSI Program submitted September 24, 1982 and June 30, 1983 and the information obtained at a public meeting in Bethesda, MD on August 31, 1983. In submittals dated July 17, 1984, August 7, 1984, August 23, 1984, August 28, 1984, and August 30, 1984 the Applicant requested relief from ASME Section XI Code requirements which have been determined to be not practical to perform. These relief requests were supported by information pursuant to 10 CFR 50.55a(a)(2)(i). Therefore, the staff evaluation consisted of reviewing these submittals and determining if relief from the Code requirements were justified. Pursuant to 10 CFR 50, Paragraph 50.55a(a)(2), the staff has allowed relief from the impractical requirements that, if implemented, would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. The detailed evaluation of relief requests is included in Appendix N to this report.

FSAR Revision 21 dated June 1983 contained a response to NRC Question 250.4 in which the Applicant provided clarification on the examination of high energy fluid system piping. Limerick Unit 1 does not use guard pipe on the high energy fluid system and all piping between containment isolation valves up to the outboard restraint will be 100% volumetrically examined during PSI and ISI.

Based on review of the Applicant's submittals, the staff has determined that the Limerick Generating Station Unit 1 PSI Program is acceptable and that the review is considered to be completed.

The initial ISI Program has not been submitted by the Applicant. The program will be evaluated after the applicable ASME Code Edition and Addenda can be determined based on Paragraph 50.55a(b) of 10 CFR Part 50, but before the first refueling outage when ISI commences.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.2.1 Instrumentation Setpoints

In the SER the NRC staff identified a concern regarding the methodology used to establish the reactor protection system setpoints. During the staff's review, it was determined that additional information would be required to confirm the applicant's conformance with the Commission's regulations relevant to the issue of protection system setpoints. The applicable regulations are: General Design Criterion 20, 10 CFR Part 50.36 and Part 50.46. Criterion 20, Protection System Functions, states that "the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety." Part 50.36 states "limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Part 50.46 specifies the performance criteria for the emergency core cooling systems. These criteria include a maximum peak cladding temperature, a maximum cladding oxidation, a maximum total amount of hydrogen generated, and requirements that core geometry remain amenable to cooling for long term decay heat removal. Guidance on acceptable methods for complying with these regulations is contained in Regulatory Guide 1.105, "Instrumentation Setpoints."

In an effort to conserve resources while providing the requested information, the applicant joined with several other BWR owners to form the Licensing Review Group (LRG) - Instrumentation Setpoint Methodology Group (ISMG). On July 14, 1983, the NRC staff met with the ISMG at their request. At this meeting the ISMG presented an outline of a setpoint methodology. In response to additional questions from the NRC staff, another meeting was held on January 31, 1984. By letter dated May 15, 1984, from T.M. Novak (NRC) to J.F. Carolan (Chairman, ISMG), the NRC staff provided its assessment of ISMG methodology. The NRC staff evaluation identified several deficiences in the methodology presented and requested that the ISMG provide additional information in response to ten specific concerns. In response to the staff's evaluation, by letter dated June 29, 1984, from J.F. Carolan to T.M. Novak, the ISMG provided an action plan for resolving the outstanding issues. By letter dated July 23, 1984, from B.J. Youngblood (NRC) to J.F. Carolan, the NRC staff accepted the proposed action plan, and by letter dated August 10, 1984, from J.S. Kemper (PECO) to A. Schwencer (NRC) the applicant committed to the work scope and schedule proposed by the action plan. The final acceptability of the protection system instrumentation setpoints will be addressed following completion of the NRC staff's review of the forthcoming additional information.

The staff concludes that there is reasonable assurance, based on staff participation in meetings with the ISMG, that the forthcoming more detailed information on the setpoint methodology being developed by this group will verify the acceptability of the proposed setpoints. In the interim, the staff finds the proposed setpoints acceptable.

7.2.2.4 Lifting of Leads to Perform Surveillance Testing

In the SER the NRC staff addressed the features in the Limerick design that provided the capability to perform surveillance tests without lifting leads. By letters dated July 25 and September 6, 1984, the applicant stated that lifting of leads will be required in order to perform a limited number of surveillance tests. For each test where lifting of leads will be required the applicant committed to follow the guidance provided in IE Information Notice 84-37, "Use of Lifted Leads and Jumpers During Maintenance or Surveillance Testing." IEN 84-37 recommends a combination of administrative controls and functional tests to verify the restoration of proper system configuration following surveillance tests.

The applicant has stated that the lifting of leads will be limited to surveillance tests that fall into one of the following categories: (1) tests that involve thermocouples, (2) tests that require the introduction of test equipment into the instrument channel being tested, (3) tests that would otherwise become unnecessarily complex, and (4) tests on systems or components for which the plant design permits no other reasonable alternative. The procedures for these tests will include instructions explicitly requiring the reconnecting of the lifted leads following the completion of the surveillance. This procedural step will be documented by a space to be initialled by the technician when the lifted leads have been reconnected. Following this, a separate verification sheet will be initialled to confirm that an independent inspection has been performed and that the lifted leads have been returned to service. Following the reconnection of the leads, functional tests will be performed to verify the restoration of proper system configuration.

Based on the results of its review, the NRC staff finds that the combination of administrative controls and functional tests meet the guidelines of IEN 84-37 and provide reasonable assurance that the instrumentation and controls will be restored to the correct configuration following testing where lifting leads is required. Therefore, the NRC staff finds the actions required acceptable.

7.2.2.5 Anticipated Transients Without Scram

In the SER the NRC staff discussed a postulated failure to scram the reactor following an anticipated transient, that is, an anticipated transient without scram (ATWS). The SER included a description of the redundant reactivity control system (RRCS) provided for ATWS mitigation. The RRCS consists of the instrumentation, controls, and actuated equipment to automatically initiate alternate rod insertion, recirculation pump trip, standby liquid control system, feedwater runback and reactor water cleanup system isolation.

By letter dated July 17, 1984, the applicant proposed to defer completing construction of the RRCS until prior to exceeding 5% of rated thermal power. Supplemental information was provided in a letter dated August 30, 1984. The applicant stated that this deferral will preclude inadvertent initiation of the standby liquid control system during the low power test period. Prior to the approach to initial criticality, all control rods are inserted and the reactor is in the shutdown mode. Between initial criticality and 5% power, the small amount of heat being generated provides the plant operators with more time than would exist at full power to manually initiate the actions which the RRCS would automatically initiate in an ATWS event.

As an interim measure until the issuance of the Commission's ATWS requirements and guidelines, the NRC staff has recommended that each Boiling Water Reactor applicant provide an automatic recirculation pump trip on high reactor pressure or low reactor vessel water level, and propose Technical Specifications to address ATWS recirculation pump trip system operabilty when the mode switch is placed in the RUN position. The mode switch is placed in the RUN position at approximately 5% power. From analysis, the staff has determined that the contribution from the recirculation pump trip in mitigating an ATWS is small at low power levels. At approximately 5% power the recirculation pumps are operating at minimum flow, comparable to natural circulation core flow. Should the pumps be tripped at this flow rate, little power reduction would result. Should an ATWS occur at or below 5% power the operators would have to perform additional actions to shut down the reactor, such as standby liquid control system initiation. This approach is acceptable to the staff because of the time available for operator action.

Based on the results of its review, the NRC staff finds the deferral of completion of the RRCS consistent with its interim position on ATWS. Therefore, the staff finds the deferral acceptable. Completion of the RRCS and its associated operability test prior to exceeding 5 percent of full power will be made a licensing condition.

On June 26, 1984 the Commission amended the Regulations to add 10 CFR 50.62 requiring each boiling water reactor to have an alternate rod injection system that is diverse from the reactor trip system from the sensor output to the final actuation device. The alternate rod injection system must have redundant scram air header exhaust valves. In addition, each boiling water reactor must have a standby liquid control system capable of injecting 86 gallons per minute of 13 weight percent sodium pentaborate solution. The standby liquid control system initiation must be automatic for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature. Further, each boiling water reactor must have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

In accordance with the provisions of 10 CFR 50.62, the applicant is required to submit a schedule for meeting the requirements not later than 180 days following the issuance by the NRC of quality assurance guidance for the ATWS mitigating system. The staff will review the design of the ATWS mitigating features for Limerick to verify conformance with 10 CFR 50.62.

7.3 Engineered Safety Feature Systems

7.3.2 Specific Findings

7.3.2.4 Restart of HPCI and RCIC on Low Water Level

In the SER the staff stated that it would review the design details when available to confirm that the current RCIC control system provides an acceptable automatic restart of the RCIC system on low water level.

By revisions to the FSAR the applicant has provided sufficient information to enable the staff to confirm the adequacy of the design. Therefore this confirmatory issue is closed.

7.4 Systems Required for Safe Shutdown

7.4.2 Specific Findings

7.4.2.2 II.K.3.22, Automatic Switchover of Reactor Core Isolation Cooling System

In the SER the staff stated that RCIC system design modifications to provide automatic transfer of the suction supply from the condensate storage tank to the suppression pool would be reviewed when the design details were made available.

By revisions to the FSAR the applicant has provided sufficient information to enable the staff to confirm the adequacy of the automatic transfer of the RCIC system suction supply. Therefore, this confirmatory issue is closed.

7.4.2.3 Remote Shutdown System

As noted in the SER the staff found that the Limerick design did not include redundant controls and display information for remote safe shutdown should the control room become uninhabitable. The staff's acceptance criteria (SRP Section 7.4) require that redundant safety-related trains be provided to shut the plant down from outside the control room. Subsequently, by letter dated July 18, 1984, the applicant committed to modify the design of the remote shutdown system to provide a redundant safety-related capability to promptly achieve and maintain hot shutdown from locations remote from the control room. The design will also include the capability for attaining subsequent cold shutdown through the use of suitable procedures. The staff had indicated that it would confirm the acceptability of the design associated with the remote shutdown system capability following receipt of the design details. To date, the applicant has not provided the required design details and, therefore, the staff cannot confirm that the remote shutdown system design fully conforms to the staff's acceptance criteria related to compliance with the requirements of GDC 19 as set forth in SRP Section 7.4.

By letter dated October 25, 1984, the applicant requested an exemption from the requirements of GDC 19 as it relates to the redundant safety-related capability to achieve hot and subsequent cold shutdown from outside the control room. To support this request, the applicant has committed (1) to provide information prior to exceeding 5% power to describe the changes necessary to upgrade the existing remote shutdown system so that it will fully comply with the subject GDC, and (2) to provide an interim (backup) remote shutdown capability using the presently installed equipment and appropriate operating procedures which will be in place prior to exceeding 5% power. The interim system will be redundant to the existing safety-related remote shutdown train which has been accepted by the staff but will not include all of the required modifications described in item (1) above. The applicant's exemption request covers two phases of operation, the first extending through initial startup and up to 5% power, and the second through the balance of the first fuel cycle to the point of startup following the first refueling outage. Justification for operation during this first phase without modification of the existing safe shutdown system has been provided by the applicant. The staff agrees that minimal decay heat removal requirements would exist prior to exceeding 5% power, and that the likelihood of loosing the control room safe shutdown capabilities and the existing safety-related remote shutdown train simultaneously is highly improbable for the short duration expected prior to exceeding 5% power.

Based on the foregoing, the staff finds that an exemption from full compliance with GDC 19 is justified for initial startup and operation up to 5% power. However, the license will be conditioned to require the applicant (1) to provide, prior to exceeding 5% power, the information on the changes to be made at the first refueling outage that will be necessary to provide a redundant safety-related method of achieving safe shutdown conditions from outside the control room, and (2) to provide a redundant remote shutdown capability using procedures and existing equipment as an interim remote shutdown system prior to operation above 5% power. The staff understands that during this interim period, these procedures may include the use of jumpering or rewiring circuits. The staff findings regarding justification for operation during the second phase will be reported in a subsequent supplement to the SER following receipt and review of the latter information.

7.5 Safety-Related Display Instrumentation

7.5.2 Specific Findings (R.G. 1.97)

Generic Letter No. 82-33 included additional clarification regarding Regulatory Guide 1.97, Revision 2 relating to the requirements for emergency response capability. On April 13, 1984 the staff requested specific information on conformance with Regulatory Guide 1.97, Revision 2. By letter dated August 16, 1984 the applicant provided the additional information requested concerning the exceptions to conformance to the Regulatory Guide. Pending the completion of the staff's review of the Limerick design for conformance to the guidance of the Regulatory Guide a condition to the license will require that modifications be completed to provide compliance with the Regulatory Guide unless the deviations are reviewed and approved by the staff prior to startup following the first refueling outage. These items as listed in the applicant's letter of August 16, 1984 are neutron flux, reactor water level, drywell sump level, drywell drain sump level, radiation level in circulating primary coolant, suppression spray flow and standby liquid control system tank level.

8 ELECTRIC POWER SYSTEMS

8.4 Other Electrical Features and Requirements for Safety

8.4.1 Physical Identification and Independence of Redundant Safety-Related Electrical Systems

Raceway Separation

The Philadelphia Electric Company, the applicant for Limerick Units 1 and 2, had committed to meet the requirements of Regulatory Guide 1.75 as stated in Section 8.1.6.1.14 of the Limerick FSAR. The applicant stated that any exceptions to the required separation criteria in R.G. 1.75 had been identified in the FSAR and justified in the Design Verification Test Report for Limerick Units 1 and 2 prepared by the applicant. The Limerick raceway design was based on the standard separation criteria contained in IEEE 384-1974 as endorsed by R.G. 1.75. Due to physical constraints, the resultant separation distance in some cases was less than the standard separation distance required by R.G. 1.75. In order to provide justification for these lesser separation distances, the applicant instituted a test program conducted by Wyle Laboratories. The test program methodology and results are documented in Wyle Test Report No. 46960-3.

The applicant submitted the test results with its associated information on the revised separation criteria dated August 16, 1984. This information was requested by our staff at the site visit of July 31, 1984.

The revised separation distances are derived from the above test results and are being implemented in the plant. This SER supplement focuses our evaluation on test results which validate the revised separation criteria.

In order to perform a test program to verify the adequacy of the raceway separation criteria, it was necessary to define the worst case electrical failure that could be postulated to occur in a raceway. The Limerick raceway separation test program was based on the following failure mode assumptions:

(1) The cable or equipment in the circuit develops a fault that is not cleared due to the postulated failure of the primary overcurrent protective devices.

(2) The fault current level is assumed to be just below the long-term trip setpoint of the next higher level (upstream) overcurrent device so that the fault is not cleared.

(3) There are no other loads on the same circuit which would cause the next higher level overcurrent device to trip.

The fault current magnitude of 660 amperes (600 amperes + 10% of uncertainty) used in the test program was based on the failure mode assumptions discussed above. This assumes that an overcurrent condition occurs on a cable between a 480 volt ac MCC and a 480 volt ac load. The primary overcurrent protective device which is a molded case breaker at the MCC is assumed to fail to trip. The next higher level (upstream) overcurrent device is the load center breaker. The fault current is assumed to be just below the long-term trip setpoint of the load center breakers which is 600 amps. This current value was used for all tests involving cables in cable raceway and was also used for tests involving cables of size #4/0 AWG or smaller in conduit. In order to select the size cable to be used for tests involving cables routed in tray or gutter, tests were performed to determine which size cable when energized with 660 amps would deliver the most intense temperature rise for the longest duration to adjacent cables. The tests showed that the 3/C #2/0 AWG cable was the worst case cable.

The Limerick motor control centers (MCC) contain Westinghouse molded case breakers which provide both overload and short circuit protection. The load centers (LC) contain ITE K600S breakers with solid state trip devices. The solid state trip devices provide increased accuracy and repeatability over conventional trip devices. The load center breakers provide both long and short time overcurrent and instantaneous short circuit protection. All breakers of the MCC and LC are tested on a periodic basis. These breakers are tested and maintained at least once every 60 months as required in the technical specifications, thereby assuring that the likelihood of two overcurrent devices in series on the same feeder line failing coincidently is extremely small.

For cables larger than #4/0 AWG in conduit, the fault current magnitude was selected as 3500 amps. This fault current magnitude is based on an overcurrent condition occurring on a 480 volt feeder from a load center to a motor control center given the failure of the load center breaker to operate. Three 1/C 750 kCM cables were chosen as the fault cables for those tests involving cables routed within conduit and energized with 3500 amps. This is the largest size cable used inside areas of the plant containing equipment important to safety and, based on the magnitude of the fault current applied, will generate the most heat.

At the completion of each cable test, the functional tests - insulation resistance test, high potential test - and overcurrent test were performed for the target cables. The target cables passed the above mentioned functional tests and overcurrent test in accordance with manufacturer's specification of the cables.

The test program with above assumptions and inputs for the target cables generated the following results:

(1) Cables sized #4/0 AWG and smaller when energized with 660 amps and routed in open cable tray did not ignite. Cables were tested in both horizontal and vertical tray configurations and did not ignite in any case. Configurations with [illegible] vertical separation between cable trays and zero separation between cable tray and enclosed raceway were tested successfully without damage.

(2) No separation was required between an enclosed raceway and another enclosed raceway or cable tray when the enclosed raceway contains cables which are #4/0 or smaller.

(3) One inch separation between an enclosed raceway and another enclosed raceway or cable tray is required when the enclosed raceway contains cables larger than #4/0 AWG.

The fault currents selected for the tests encompass the conditions which can result from failures of the high speed overcurrent protective relaying on the feeder line. These features will cause either upstream protective relaying operation or rapid cable failure, thereby preventing long-term heat generation to ignition. The postulated fault current, 660 amperes, on the cable size smaller than or equal to #4/0 AWG is reasonably adequate based on the design of the overcurrent protective relaying on the feeder and the upstream power supplies. The staff has concluded that the test program with above assumption and input are acceptable, and so are the test results.

The staff has reviewed the application of the test results to the raceway separation criteria contained in Section 2.0 of Limerick Drawing 8031-E-1406, Attachment 2 of the letter dated August 16, 1984. Based on the staff's review of the applicant's design of the overcurrent protective systems and its test program and test results, the staff finds the justification for the deviations from the criteria of R.G. 1.75 and the revised separation criteria acceptable.

The staff's conclusion is based on physical separation as it pertains to electrical fires initiated by electrical faults, occurring as a single failure during a design basis event and does not pertain to nor modify the 10 CFR 50 Appendix R criteria which addresses exposure fires.

Terminal Block and Panel Meter Separation

The applicant revised FSAR Section 8.1.6.1.14.b.9(5), to include justification for terminal blocks and the panel meters exceptions from separation criteria as recommended by RG 1.75. These two items are additional to the indicating lamps and isolation relays identified in the Design Verification Test Report on Internal Panel Control Wiring Separation (Report No. 48503), which the NRC staff previously accepted as stated in Section 8.4.1 of the SER.

The applicant submitted Wyle Test Report No. 46960-4, Electrical Separation Verification Testing on Terminal Blocks and Panel Meters, for the staff's review. This test report provides the justification for the internal panel separation criteria which permits termination of redundant Class 1E control circuits on adjacent terminal points on a common terminal block and permits mounting panel meters side by side with no physical separation. The applicant performed a series of tests during 1981 to determine the separation criteria to be applied for internal panel control wiring at Limerick. IEEE Standard 384-1974, endorsed by RG 1.75, allows the use of less than six inches of spatial separation if a lesser distance can be shown to be adequate by analysis and/or test. The report documents the testing performed on the terminal blocks and the panel meters to justify the proposed revised separation criteria that were implemented in the wiring of the Limerick control panel. This supplement to the SER presents our evaluation of test results which validate the revised separation criteria for these two items.

a. Terminal Blocks

The majority of the cables terminating on the terminal blocks to be tested are Size 14 AWG. The test showed that for Size 14 AWG conductors, the maximum current which was carried continuously was 90 amperes and that the conductors failed open when energized with 100 amperes. In order to verify that adjacent terminal points on a terminal block provide adequate separation, it must be demonstrated that on an overcurrent condition the Size 14 AWG conductors fail prior to any degradation of the terminal block. To verify this, a terminal point must be capable of carrying 100 amperes continuously without degradation of a circuit on an adjacent terminal point.

The test demonstrated that with 100 amps of ac current applied through a terminal point for 20 minutes, there was neither interference with nor interruption of a 10-amp ac current signal applied to an adjacent terminal point. In addition, with a difference in potential of 4000 Vac applied between two adjacent terminal points, there was no evidence of insulation breakdown or flashover.

It is therefore concluded that a single-point terminal barrier provides adequate electrical separation, during worst electrical separation fault conditions, between redundant Class 1E electrical systems or between a Class 1E and a non-Class 1E electrical system.

At the completion of each terminal block test, the insulation resistance test, overcurrent test and voltage breakdown test were performed for the terminal block and a circuit on an adjacent terminal point. The terminal block and the circuit on an adjacent point passed these tests by meeting the acceptance criteria of the above mentioned tests.

b. Panel Meters

The Limerick design includes panel meters mounted side by side with no physical separation. The internal circuitry of the GE Model 180 Edgewise Panel Meter requires performance of both overvoltage and overcurrent tests to demonstrate the adequacy of separation.

A condition can be postulated that a cable connected to a panel meter through the potential transformer 480/120 Vac could become energized with 480 Vac due to the transformer internal fault (short circuit between the 480 and 120 volts). Therefore, to demonstrate the adequacy of meter separation, overvoltage tests applying 480 Vac minimum to the meter must be performed.

It was demonstrated by the test that the application of overvoltage of 600 Vac, and 5 amperes of fault current (250 times the maximum meter input) to a panel meter would not in any way affect the indication of another panel meter adjacent to and in contact with first meter. It is therefore concluded that adequate physical and electrical separation exists when two panel meters are mounted adjacent to each other without any physical separation between them during the application of any credible electrical fault at Limerick.

At the completion of each panel meter test, the operability test, the accuracy test, and overcurrent/overvoltage test were performed for the other meter as recommended by the manufacturer's specification and the other meter exceeded the acceptance criteria.

Based on the staff's review of the test program and test results, the staff finds the justification for the deviations from the criteria of RG 1.75 and the revised separation criteria for the terminal blocks and the panel meters acceptable.

Our conclusion is based on physical separation and electrical separation as it pertains to damage from electrical faults and electrical interferences initiated by electrical faults, occurring as a single failure during a design basis event and does not pertain to nor modify the 10 CFR 50 Appendix R criteria which addresses exposure fires.

9 AUXILIARY SYSTEMS

9.2 Water Systems

9.2.1 Emergency Service Water System

The staff noted in Section 9.2.1 of the SER that Unit 1 takes credit for redundancy by using Unit 2 equipment. The staff further indicated that the license would be conditioned to require that the Unit 1 boundary be defined, for security purposes, to contain all necessary emergency service water pumps, related piping and all isolation valves. The applicant has addressed this subject in letters dated May 25, August 1 and August 24, 1984 and in specific revisions to the security plan. Accordingly, this staff requirement for definition of the Unit 1 boundary has been met by specification of the boundary in the security plan. Implementation of the physical security plan is addressed by a specific condition to the operating license.

9.2.2 RHR Service Water System

The staff noted in Section 9.2.2 of the SER that the license would be conditioned to require that the Unit 1 boundary be defined, for security purposes, to contain all necessary RHR service water pumps, related piping and isolation valves. The applicant has addressed this subject in letters dated May 25, August 1 and August 24, 1984, and in specific revisions to the security plan. Accordingly, this staff requirement for definition of the Unit 1 boundary has been met by specification of the boundary in the security plan. Implementation of the physical security plan is addressed by a specific condition to the operating license.

9.2.5 Ultimate Heat Sink

We stated in our SER that the Limerick Ultimate Heat Sink (UHS) did not meet the requirements of General Design Criteria (GDC) 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Missile Design Bases." The UHS consists of a spray pond and the Schuylkill River. As stated in the SER, the spray pond is designed to withstand earthquakes, floods, and freezing. However, neither the the spray network nor the Schuylkill River pumphouse are designed to withstand missiles generated by natural phenomena.

In response to the requirement for protection against missiles, the applicant chose to provide a PRA evaluation. The applicant's PRA was submitted in March 1984 and additional information was provided in submittals dated September 4 and 11, 1984. In the PRA evaluation, the applicant determined the probability of tornado, hurricane and straight wind-borne missile damage to the spray pond and the cooling towers. We and our consultant have reviewed these submittals.

Our consultant's technical evaluation report (TER) of the applicant's PRA considered the validity and conservatism of the approach, assumptions and data used in the applicant's analysis. Also included in the TER is an assessment of the correctness of the results obtained in the study.

We have reviewed our consultant's TER and concur with the findings that the estimate of the probability of exceeding 10 CFR Part 100 limits owing to wind effects on the spray pond and the cooling towers will not exceed 1E-6 per year. The consultant's TER is a part of our SER and is included as Appendix O of this report. In the unlikely event that wind effects prevent use of the cooling towers and spray network for shutdown of Unit 1, there are alternate cooling modes available to provide cooling while repairs of the spray pond are being made to return at least one spray network to service. The applicant described these alternate measures in a letter to Mr. A. Schwencer dated October 19, 1984. The applicant stated that upon loss of the spray network and cooling towers, the spray pond will be operated in a cooling pond mode until the temperature of the water reaches the design limit of 95°F. In this mode, water will be returned to the pond via the winter bypass line to promote thermal mixing and minimize the likelihood of recirculation. Under design basis conditions of initial pond temperature and meteorology, it would take approximately 6 hours for the pond to reach its 95°F limit. Under average conditions, it would take approximately 10 hours to reach this limit. Both numbers (6/10 hours) are for two unit, full power operation. For single unit operation, these times would be approximately 12 hours and 20 hours respectively. The heat rejection rate can be further reduced by depressurizing the reactor at a slower rate than 100°F/hr assumed in the design basis analysis.

When the pond reaches the design temperature limit, the sluice gates between the spray pond pumphouse wet wells and the spray pond will be closed. Water will then be released from the cooling tower basins into the wet wells and pumped through the plant to service the required heat loads. The water will be returned to the spray pond and will be allowed to discharge over the blowdown weir and storm spillway. The two cooling tower basins contain a total of 14 million gallons. The applicant assumed that only one half of this volume of water is available, which is sufficient water to provide makeup for the emergency service water (ESW) and RHR service water (RHRSW) pumps, operating in a once through mode, for an additional 4 hours. The applicant stated that if the cooling towers fail due to tornado and hurricane winds, the debris would be expected to fall into the basins of the towers in large chunks which would not block the drainage of water from the basins. In the unlikely event that the cooling tower basin walls have failed due to tornado missiles, the additional time of four hours would not be available.

Sufficient makeup water can be supplied to the cooling tower basins to sustain continuous operation in this mode, if the Schuylkill River makeup pumphouse is not damaged by the tornado. The pumphouse is located approximately 1500 ft. from the nearest cooling tower, making it unlikely that the pumphouse would be damaged by a tornado which would also compromise the spray pond networks and the cooling towers. This pumphouse is powered from the 2300-Volt plant services switchgear. This switchgear can be fed using offsite power from either of the two plant substations via underground lines. The two substations are approximately 2000 ft. apart, making it highly unlikely that both substations would be disabled by a tornado which would also compromise the spray pond networks and the cooling towers.

If existing sources of makeup cannot be made available, makeup will be provided using available portable pumps of required size and capacity to pump water from the Schuylkill River to the spray pond pumphouse wet wells. The water would be pumped via a tie-in to the existing underground water pipeline which runs from the Schuylkill River Intake Pumphouse to the cooling tower basins. It would then flow via gravity to the pump pits. If a tie-in to the existing pipeline is not possible, then the water would be pumped directly to the wet well through temporary lines. The portable pumps which will be used are either PECo owned pumps or rental pumps. The required pumps will be verified to be available prior to exceeding 5% power and yearly thereafter.

The repair work on the damaged spray networks will begin immediately, utilizing materials, equipment and personnel which have been verified to be available. A plant procedure (which will be approved and implemented prior to exceeding 5% power) will govern such repair activities. Procedure verification will be made each year. Since the repair procedure and the verification of materials, equipment, personnel, and portable pumps will not be completed until Unit 1 is ready to exceed 5% power, the applicant requested a schedular exemption to the requirements of GDC 2 and 4, as they relate to the protection of the UHS from the effects of tornado missiles. This request was submitted in a letter to Mr. H. Denton dated October 19, 1984.

The applicant stated, as their basis for the exemption request, that during the period of operation before exceeding 5% power, it is extremely unlikely that tornado missile damage to the spray networks would occur. Even if the heat removal capability of the cooling towers and spray networks were compromised by tornado missile effects, use of the cooling tower basins and/or UHS in a "cooling pond type" mode would allow substantial time for spray network repair. Under design meteorology, it would take approximately 5 days for the pond to reach its 95°F limit.

In the remote possibility that the heat removal capability of the spray pond networks and the cooling towers is compromised, and that repairs cannot be completed before the design temperature of the spray pond is reached, a once-through mode of cooling can be implemented. In this mode of operation, water from the cooling tower basins is supplied to the spray pond pumphouse wet pits, ESW and RHRSW will pump this water through the plant, the water is returned to the spray pond and is allowed to discharge over the blowdown weir and storm spillway. Sufficient makeup water can be supplied to the cooling tower basins to sustain continuous operation in this mode from the Schuylkill River. This provides sufficient time to effect the repairs on any one of the four networks such that sufficient heat removal capability can be restored without the existence of specific procedures. Specific procedures for such repairs will be completed prior to exceeding 5% power.

Based upon the applicant's October 19, 1984 letter, we have determined that an exemption from the requirements of GDC 2 and 4 is acceptable, for criticality and low power testing not to exceed 5% power, because it does not cause any undue risk to the health and safety of the public.

Based on the above, we conclude that the applicant has satisfactorily demonstrated compliance with General Design Criteria 2, and 4, with respect to protection against natural phenomena and missiles for Unit 1 operation, and is, therefore, acceptable. Thus, Unit 1 meets the acceptance criteria of SRP 9.2.5.

9.2.8 Control Structure Chilled Water System

The staff noted in Section 9.2.8 of the SER that the license would be conditioned to require that the Unit 1 boundary be defined, for security purposes, to contain all necessary control structure chilled water system pumps, piping and isolation valves. The applicant addressed this subject in a letter dated August 1, 1984, noting that the CSCW system is located entirely within the Unit 1 security boundary and that no special security provisions would be required for this system. Accordingly, this staff requirement has been met by implementation of the security plan. Implementation of the physical security plan is addressed by a specific condition to the operating license.

9.4 Heating, Ventilation and Air Conditioning Systems

The staff stated in Section 9.4.1 of the SER that the applicant must consider the control structure ventilation system equipment as Unit 1 equipment for security purposes. The applicant has addressed this subject in letters dated May 1, August 1 and August 24, 1984 and in specific revisions to the security plan. Accordingly, this staff requirement has been met by implementation of the security plan. Implementation of the physical security plan is addressed by a specific condition to the operating license.

10 STEAM AND POWER CONVERSION SYSTEM

10.2 Turbine Generator

By letter dated October 12, 1984, the applicant has proposed to change the FSAR requirement for testing the turbine main steam control valves from once per 7 days to once per 31 days. The basis for this proposal is a General Electric Technical Information Letter (TIL) dated May 22, 1984, and numbered 969. In TIL 969, General Electric states that "turbine steam inlet valve reliability and testing intervals are no longer the major contributing factors in determining hypothetical turbine missiles," and that "the overall probability of a hypothetical turbine missile is therefore increased only a negligible amount by increasing the test interval of the valves." The above is based on accumulated in-service experience at nuclear plants over 24 years. As a consequence of this experience, General Electric, in TIL 969, recommends changing the test frequency for turbine control valves from weekly to monthly. The staff concurs with the applicant's position. Therefore, the proposal to change the testing frequency for the turbine control valves from once per 7 days to once per 31 days is acceptable.

By letter dated October 12, 1984, the applicant has proposed that turbine main steam valve movement during testing be monitored using remote position indicators, and that the FSAR surveillance requirement to visually observe this movement every 31 days be deleted. The basis for the deletion is that the physical location of the turbine main steam valves (high pressure stop and control, and combined intermediate and extraction) and other equipment in the turbine building makes it necessary for an observer to enter a plant radiation zone 5 in order to visually observe valve movement. Since all turbine main steam valves are equipped with reliable and redundant remote position indicators, there is no rationale for exposing personnel to high radiation simply for visual confirmation of valve motion. The staff concurs with the applicant's position. Therefore, the surveillance requirement to visually observe turbine main steam valve movement every 31 days can be deleted.

11 RADIOACTIVE WASTE MANAGEMENT

11.4 Solid Waste Management System

In the SER the staff concluded that the applicant's proposed solid radwaste system is acceptable on the condition that the applicant provide a process control program. In a letter dated August 23, 1984, the applicant submitted a process control program with a stipulation that a final process control program will be submitted upon analysis of the results of preoperational testing of the solid radwaste system.

The process control program provides guidance and boundary conditions for preparation of specific procedures for processing, sampling, analysis, packaging and shipment of solid radwaste in accordance with State and Federal regulatory requirements including Technical Specification 3.11.3. The centrifuge setting for resin dewatering, however, will be verified or adjusted as required during preoperational testing.

Based on our review of the process control program submitted with the August 23, 1984 letter, we find the process control program to be acceptable pending submittal and acceptance of the operating points in the final process control program as verified or adjusted based on the preoperational testing of the solid radwaste system.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.1 System Description

In letters dated July 25, 1984 and August 17, 1984, the applicant provided final design details of the wide range noble gas effluent monitor for post-accident releases and provisions for sampling and analysis of gaseous effluents for post-accident releases of iodines and particulates. Included was a description of calculational methods to be used for converting noble gas effluent monitor instrument readings to release rates per unit time. Based on our review, we conclude that the final design of this instrumentation is consistent with the requirements of NUREG-0737, Item II.F.1, Attachments 1 and 2.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Operation

13.1.2.2 Personnel Qualifications

Shift Advisor Program

In Section 13.1.2.2 of the SER the staff discussed the issue of ensuring that adequate shift personnel with substantive previous commercial BWR startup and operating experience were provided. By letter dated May 31, 1984, the staff requested additional information on this subject.

Shift Advisor Program. The applicant's response by letter dated August 21, 1984, submitted information regarding its Shift Advisor Program. We have reviewed this program for conformance to the guidelines for Shift Advisors proposed to the Commission by the Industry Working Group on February 24, 1984, and accepted with some clarifications by the Commission on June 14, 1984. In conducting our review, we have also used information and additional thoughts regarding qualifications and training of Shift Advisors developed during our review of Shift Advisor Programs at other utilities.

On February 24, 1984, an Industry Working Group representing utilities that had nuclear plants under construction or ready for operation made a proposal to the Commission on the amount of previous operating experience considered to be the minimum desirable on each shift, and how that experience could be obtained. On June 14, 1984, the Commission accepted the industry proposal with certain clarifications. Information regarding the Commission action was forwarded to the industry as Generic Letter 84-16, dated June 27, 1984. The ultimate objective is that at time of fuel load, each operating shift will have at least one senior operator who has had a minimum of six months of previous hot operation experience on a similar type plant, including at least six weeks of experience above 20% power, and including startup and shutdown experience. However, for plants in the late stages of licensing when there is not sufficient time to provide adequate hot experience to plant personnel, the use of experienced advisors to each of the operating shifts is acceptable. The minimum qualifications prescribed for the Shift Advisors are four years of power plant experience, including two years of nuclear plant experience, with a minimum of one year experience as a licensed senior operator or operator (if found suitably qualified) on a large, commercial nuclear plant of the same type. Each advisor is to be trained on the systems, procedures, and technical specifications of the plant to which they are to provide advice, and certified to the NRC as being qualified to act as Shift Advisors. Limerick falls within the group of plants eligible to use advisors to provide experienced advice to the operating shifts.

By letter of August 28, 1984, the applicant has updated the information contained in the August 21, 1984, submittal. The applicant has advised us that only one of the five operating shifts will require the use of a Shift Advisor.

We have reviewed the procedure which defines the responsibilities and duties of the Shift Advisors and, with the following recommendation, conclude it meets the guidelines adopted by the Commission. We recommend that the Shift Advisors participate in requalification training which enables them to be cognizant of facility design, procedure, and license changes. In addition, the Shift Advisors should participate in scheduled requalification simulator exercises and shift training when appropriate.

We have reviewed the requalifications of the two prospective Shift Advisors. Both candidates held SRO licenses at the applicant's Peach Bottom 2 and 3 Station and performed on-shift duties as Senior Operators exceeding the period contained in industry/Commission guidelines.

We have also reviewed the training administered to the Shift Advisor candidates. The training consists of instructions in Limerick systems, procedures, technical specifications, plant tours, and exercises using the Limerick simulator. The training period is six weeks long, during which weekly examinations are administered. Final written and oral examinations are given at the end of the training period. The material covered during the training period is appropriate to meet the industry/Commission guidelines. The training period was scheduled to end by September 28, 1984, and the applicant has committed to furnish us copies of the written and oral questions as well as the examination results. We have also been advised that a Region I Operator Licensing examiner will monitor the examinations.

With regard to training the operating shift crews on the role of the Shift Advisors, the applicant has stated that a memorandum describing the responsibilities and authority of the Shift Advisor will be discussed with operating shift personnel. The applicant should provide the Commission the date that this task has been accomplished.

The applicant has also stated that the company medical department will examine the Shift Advisors or review existing medical records in the light of their duties and responsibilities in order to assure that the individuals are qualified. We find this commitment acceptable.

With regard to a performance evaluation of the Shift Advisors, the applicant has submitted a performance evaluation form with criteria. We find the criteria selected are among the best we have reviewed to date. However, the applicant has not stated the frequency of evaluation. We recommend the applicant perform monthly evaluations of the Shift Advisors. We believe that the data developed, from the limited number of Shift Advisors, will be useful in determining the effectiveness of the Shift Advisor Program.

Overall, we find the applicant's program for providing operating experience on each shift to be in accordance with the Commission's guidelines and, therefore, acceptable. The operating license will be conditioned to ensure that this operating experience is provided.

13.3 Emergency Planning

13.3.1 Introduction

The Philadelphia Electric Company submitted the Emergency Plan for the Limerick Generating Station (LGS) as part of the FSAR. The plan was reviewed against the requirements of 10 CFR 50.33 and 50.47, Appendix E to 10 CFR 50, and Regulatory Guide 1.101, Revision 2, which endorses the evaluation criteria in NUREG-0654/FEMA-REP-1, Revision 1, entitled "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," dated November 1980. The review was conducted in accordance with SRP 13.3 (NUREG-0800).

The review identified numerous deficiencies which resulted in requests for additional information which were transmitted to the applicant in a letter dated September 15, 1982 and clarified in subsequent telephone conversations. The applicant responded with revisions to the Emergency Plan the latest of which is Revision 10 dated August 31, 1984. The applicant also submitted emergency plan implementing procedures, the latest submittal being dated August 7, 1984. Additional commitments were provided to the NRC by the applicant in a letter dated September 18, 1984.

During the period of June 11-22, 1984, the NRC conducted an onsite appraisal of the emergency preparedness program at the Limerick Generating Station. The objective of the appraisal was to evaluate the adequacy and effectiveness of the applicant's onsite emergency preparedness program and to identify areas of weakness that needed to be strengthened. The appraisal team reviewed selected procedures and representative records, inspected emergency facilities and equipment, observed activities and interviewed personnel. The findings of the emergency preparedness appraisal are contained in an NRC inspection report (50-352/84-18) dated August 14, 1984. The findings of the appraisal indicated that certain corrective actions are required in the applicant's emergency preparedness program. The applicant responded to the appraisal report findings in letters dated September 7 and 27, 1984. The following evaluation report is based on the NRC staff's review and evaluation of the applicant's responses to the appraisal findings as well as the information submitted by the applicant referred to above. Section 13.3.2 of this report lists each standard of 10 CFR 50.47(b) followed by a summary of the applicable portions of the applicant's Emergency Plan as they apply to the standard. Section 13.3.3 of this report provides the staff's conclusions.

13.3.2 Evaluation of the Emergency Plan

13.3.2.1 Assignment of Responsibility (Organizational Control)

Planning Standard

Primary responsibilities for emergency response by the nuclear facility licensee, and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

Emergency Plan Evaluation

The LGS Plan identifies the following State, local, Federal and private sector organizations that are intended to be part of the overall emergency response organization within the emergency planning zones:

a. Pennsylvania Emergency Management Agency (PEMA)

b. Pennsylvania Department of Environmental Resources/Bureau of Radiation Protection (BRP)

c. Pennsylvania State Police (only for security related actions)

d. Montgomery County Office of Emergency Preparedness and Medical Services

e. Chester County Department of Emergency Services

f. Berks County Emergency Management Agency

g. NRC Region I

h. Department of Energy, Brookhaven Area Office

i. State of Maryland

j. State of New Jersey

k. State of Delaware.

The identified industry resources are the Institute of Nuclear Power Operations, the General Electric Company and Bechtel Corporation.

When an emergency occurs in the station, the Interim Emergency Director is responsible for taking immediate action to safeguard personnel and equipment. Utilizing the implementing procedures, the Interim Emergency Director who initially is the Shift Superintendent notifies government agencies and activates the necessary portions of the emergency response organization consistent with the degree of severity of the emergency. The Station Superintendent, or in his absence the Assistant Station Superintendent, assumes the duties of the Emergency Director when onsite.

There is a 24 hour per day communication linkage capability between the facility and Federal, State and local response agencies and organizations to assure rapid transmittal of accurate notification information and emergency assessment data.

The applicant's concept of operations is described and the relationship of the applicant's emergency organization to the total emergency response effort is shown in Figure 3-1 of the Plan. Block diagrams illustrate the information flow, emergency notification, responsibility matrix, and initial and recovery phase organizations.

Written agreements are included to verify assistance arrangements between the plant and other support organizations to provide for radiological support, medical assistance, medical transportation and fire protection.

Figure 5-5 of the LGS Plan is a personnel and facilities planning basis summary which shows the staffing for prolonged emergency operations. The Corporate Emergency Support Officer is responsible for assuring continuity of resources and has the authority, management ability, technical knowledge, and procurement authority to commit corporate resources and to manage these support functions.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50 Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.2 Onsite Emergency Organization

Planning Standard

On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified.

Emergency Plan Evaluation

Figure 5-1 of the LGS Emergency Plan describes the normal station organization including shift operations during routine operation. Figure 5-2 describes the composition of emergency teams, both on-shift and when augmented by plant staff.

The Interim Emergency Director is the Shift Superintendent. The alternate is the Shift Supervisor. These positions are filled 24 hours per day on rotating shifts. It is Philadelphia Electric Company policy that the assessment, declaration of emergency conditions, immediate response, activation of the emergency organization, offsite notifications, recommendations for offsite protective actions, and implementation of onsite corrective and protective measures, as described in the Plan, are the responsibility of the Interim Emergency Director until relieved of those responsibilities by the Emergency Director. The duties of the Interim Emergency Director include:

a. Verify the existence of an emergency, classify the emergency, and decide that notifications are to be made.

b. Remain in the Control Room area and maintain authority to direct actions during the incident.

c. Notify plant personnel and activate appropriate portions of the emergency organization.

d. Notify offsite organizations and agencies.

e. Verify proper operation of plant systems and monitors.

f. Perform assessment actions and monitor the effects of the emergency.

g. Provide status and assessment information to appropriate offsite emergency response agencies such as NRC, PEMA, and BRP.

h. Provide recommendations for protective actions directly to Commonwealth officials or, if warranted in a General Emergency, to County officials. Protective action recommendations will be determined in accordance with applicable LGS procedures and Commonwealth plans such that in a General Emergency direct recommendations will be provided.

i. Implement the provisions of the Plan and applicable plant procedures. Regardless of existing plans, the judgement of the Interim Emergency Director plays a vital role in any emergency, and may in some cases take precedence over previously proposed actions.

j. Initiate protective measures onsite. The safety and well-being of station personnel are the responsibility of the Interim Emergency Director. No planned radiation exposures in excess of normal station administrative guides are permitted without the authorization of the Interim Emergency Director.

k. Strictly enforce existing procedures regarding Control Room access and formality in order to prevent crowding and to ensure that the line of command remains clear.

Items a,b,h,i,j,k are not to be delegated to other members or segments of the emergency organization. The remaining items may be carried out by other emergency personnel under the direction of the Interim Emergency Director.

The Emergency Director is the Station Superintendent. The alternate is the Assistant Station Superintendent. The Emergency Director assumes his duties as soon as onsite and thoroughly cognizant of the situation. The Emergency Director will normally report to the Technical Support Center (TSC) but has the prerogative of going to the Control Room. The Emergency Director has direct responsibility for plant operations and reports to the Site Emergency Coordinator, if this functional position is activated.

The Site Emergency Coordinator is the Superintendent - Nuclear Generation Division. The primary alternate is the Superintendent-Nuclear Services Division and the secondary alternate is the Station Superintendent-Peach Bottom Atomic Power Station. The Site Emergency Coordinator, when notified at a Site or General Emergency, normally goes to the Emergency Operation Facility (EOF). The Site Emergency Coordinator assumes overall control of the emergency organization from the (Interim) Emergency Director.

Specific assignment to emergency tasks is shown in Figure 5-2 of the Emergency Plan. These assignments cover the emergency functions shown in Table B-1 of NUREG-0654 (Table 2 of Supplement 1 to NUREG-0737) regarding the minimum staffing requirements for nuclear power plant emergencies. The applicant has conducted surveys of the normal one way travel time from home to work of employees. These surveys show that shift staff augmentation can be accomplished to meet the objectives of the thirty and sixty minute response times in Table B-1 of NUREG-0654.

Figures 3-1 and 3-2 of the Plan show the interfaces between and among the onsite functional areas of emergency activity, offsite licensee support, and local and State government response agencies. These figures include the onsite Technical Support Center, the Operational Support Center, and the Emergency Operations Facility.

Figure 5-4 diagrams the Philadelphia Electric Company (PECO) corporate support functions which are available to augment and assist the plant as necessary to cope with emergency conditions. The central location for activating and coordinating these support functions is the PECO Headquarters Emergency Support Center on the 7th floor of the Philadelphia Electric headquarters building at 2301 Market Street, Philadelphia. If conditions of the emergency indicate the need, specific support functions would be moved to the plant site.

The plan includes written agreements with the following local support sources:

Radiation Management Corporation

Pottstown Memorial Medical Center

Limerick Fire Company

Goodwill Ambulance Service

Linfield Fire Company

Dr. Arthur Mann of Pottstown, Pa.

Dr. Charles W. Delp of Boyertown, Pa.

These letters describe available support services and the limits on the actions of the persons performing those services.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50 Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.3 Emergency Response Support and Resources

Planning Standard

Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

Emergency Plan Evaluation

All the resources of the Federal agencies appropriate to the emergency condition would be made available in accordance with the Federal Master Plan. This plan and the resources behind it can be activated by LGS either by notification through the NRC or the State, or by direct notification. This effort could involve manpower and equipment for extensive plume measurement, including aerial monitoring and tracking, and sampling and analyses of ingestion pathway media. The Federal Radiological Monitoring and Assessment Plan (FRMAP) team is located at Brookhaven National Laboratory in Upton, Long Island. The (Interim) Emergency Director is authorized to call for this assistance. Office space and communications are available in the EOF to support the Federal response.

Section 5.3.3.4 of the LGS Emergency Plan states that the NRC, Region I, will dispatch personnel to the Technical Support Center and the Emergency Operations Facility for accident evaluation. The Region I office is located in King of Prussia, Pa., less than one hour travel time by auto from Limerick.

The laboratories of Brookhaven Laboratory, the Pennsylvania Bureau of Radiation Protection (BRP) and the Canberra Radiation Management Corporation are available to provide radiological analysis services. Contact has also been made for support in an emergency with the Institue of Nuclear Power operations, General Electric, Bechtel, and with neighboring nuclear power plant licensees.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50 Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.4 Emergency Classification System

Planning Standard

A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

Emergency Plan Evaluation

The applicant's emergency plan establishes an emergency classification scheme in accordance with that set forth in Appendix 1 to NUREG-0654 (Regulatory Guide 1.101, Rev. 2). The four classes of emergency are: Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency.

Observable and measurable Emergency Action Levels (EALs) have been established which, if exceeded, will initiate each emergency class consistent with the criteria of Appendix 1 to NUREG-0654.

The LGS Plan includes a list of the postulated accidents with a potential for offsite consequences analyzed in Chapter 15 of the FSAR, with a correlation for each accident to one of the four emergency classes. The onsite monitoring systems for classifying emergencies are identified in Emergency Plan Implementation Procedure EP-101, "Classification of Emergencies."

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50 Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.5 Notification Methods and Procedures

Planning Standard

Procedures have been established for notification, by the licensee of State and local response organizations and for notification of emergency personnel by all response organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.

Emergency Plan Evaluation

Section 6.1 of the LGS Emergency Plan states that an emergency classified as an Unusual Event or greater is reported to the Pennsylvania Emergency Management Agency (PEMA) and to Montgomery County and that PEMA notifies Chester and Berks Counties to ensure that all of these agencies are well informed and can address concerns of the public, government officials and news media. The notification shall be within about 15 minutes from the time at which the operators recognize events have occurred which make declaration of an emergency class appropriate.

Appendix F of the LGS Emergency Plan contains the message formats for the initial notification messages for each of the four classes of emergency. Section 6.2 of the LGS Emergency Plan states that follow-up messages will include the following as applicable to the emergency:

1. Location of incident and date/time of occurrence;

2. Identification of personnel at communication points;

3. Emergency class;

4. Actual or potential radioactive release type (airborne, waterborne, surfacr spill) and duration;

5. Estimate of quantity of radioactive material released or being released and the points and heights of release;

6. Chemical and physical form of released material, including estimates of the relative quantities and concentrations of noble gases, iodines and particulates;

7. Meteorological conditions at appropriate levels (wind speed, direction, indicator of stability, precipitation, if any);

8. Actual or projected dose rates and integrated dose rates at the site boundary and at other distances from the plant;

9. Projections of integrated doses for affected sectors and distances (2, 5 and 10 miles);

10. Estimates of surface radioactive contamination;

11. Status of emergency response actions;

12. Recommended emergency actions, including evaluation of protective action options;

13. Requests for assistance;

14. Prognosis for worsening or termination of the event based upon plant information.

Section 7.2.13 describes the system of sirens located throughout the 10 mile EPZ for alerting the public to tune to the Emergency Broadcast System for further information. The design of the system is based on Appendix 3 of NUREG-0654. The sirens are controlled by digital encoded radio signal on a county by county basis. The county controls the activation of the sirens from its Emergency Operations Center through a central transmitter located at Limerick. The risk counties may be notified by PEMA or directly by plant personnel.

The capability of the counties to make a prompt protective action decision, activate the sirens and notify the public within about 15 minutes after being notified by the plant operators of an emergency requiring urgent action will be verified during the course of the review of offsite plans and preparedness by FEMA. The FEMA report of the July 25, 1984 exercise dated September 25, 1984 identified excessive time to develop protective action recommendations and activate the alert and notification system as a significant deficiency in offsite preparedness. This matter is considered by the NRC staff to be an open item requiring resolution prior to operation above 5% of rated power and will be further evaluated upon receipt of FEMA's supplemental interim finding report on preparedness. The NRC staff has verified by direct observation during the July 25, 1984 exercise that the applicant has the capability in the Technical Support Center to activate the sirens and to inform local radio stations of the need for a warning broadcast.

The siren alert system consists of a total of 165 sirens. As of October 3, 1984, 138 sirens had been installed and operationally tested, 25 sirens were being installed, and sites for the remaining two sirens were being acquired. The applicant projected that the entire system would be installed and operational by mid-October, 1984. The staff will require verification that a viable siren alert system consisting of the majority of sirens is installed and operationally tested prior to fuel load and that the entire system is in place and tested prior to operation above 5% of rated power.

Appendix G of LGS Emergency Plan is the Corporate Communications Plan which provides for written press information and for the use of prepared statements in response to telephone inquiries. The information noted above, listed in Section 6.2 of LGS Emergency Plan, provides supporting information for the use of governmental agencies in their drafting messages for the public.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50 Appendix E, and the guidance criteria of NUREG-0654 with the exception of the prompt decision-making capability of offsite officials noted above to be resolved prior to exceeding 5% of rated power.

13.3.2.6 Emergency Communications

Planning Standard

Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

Emergency Plan Evaluation

Section 7.2 of the LGS Emergency Plan describes the emergency communications network for notifying and coordinating activities with onsite and offsite emergency response organizations. The system is designed to provide secure, redundant and diverse communications to all essential onsite and offsite locations during normal and accident conditions. Onsite systems are comprised of an intra-plant public address system, a private automatic branch exchange telephone system (PABX), and an intra-plant maintenance telephone system which is part part of the PABX system and consists of telephone jacks into which portable dial telephone sets may be plugged. Radio capability is provided between the LGS Control Room, PECO Headquarters and other PECO generating stations, and the Montgomery County Office of Emergency Preparedness and Medical Services, as well as between the TSC and EOF and offsite monitoring teams.

Figure 7-2 of LGS Emergency Plan shows emergency communication links for the Control Room, OSC, TSC, EOF, Emergency News Center, PECO Headquarters Support Center, PEMA, Montgomery County, Berks County, Chester County, Pennsylvania BRP and the NRC via a dedicated switch which provides for rapid and reliable dial and conferencing capability. Leased tie lines, supplemented by private microwave lines, link LGS, the EOF, PECO Headquarters Support Center and the Emergency News Center. Two circuits are dedicated to NRC communications; the Health Physics Network and the Emergency Notification System.

The onsite evacuation alarm consists of sirens and a public address system, and the river warning system which can transmit warning instructions through broadcast speakers located adjacent to the river. Normal telecommunication channels will be used in notifying the ambulance service dispatch center. The ambulance is capable of radio communications with the hospital while enroute with a patient.

Section 8 of the LGS Emergency Plan specifies the frequency of communications testing which is in compliance with 10 CFR 50 Appendix E, Section IV.E.9 and Criterion N.2.a of NUREG-0654.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50 Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.7 Public Education and Information

Planning Standard

Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.

Emergency Plan Evaluation

Philadelphia Electric Company maintains an Information Center near the Limerick site which is open to the public. Organized visits have been made to date by school classes, and social, civic and technical groups. A Company speakers' bureau is available for invitations from various organizations to discuss and debate topics germane to nuclear power and radiation protection.

Company personnel have participated in radiation training of local county emergency personnel. The applicant also offers an annual training program to news media personnel. Topics include explanations of the workings of nuclear reactors, radiation hazards and radiation protection, and news release procedures. The seminar is developed and conducted by a Company consultant. Information kits are available to news media personnel, which include information on emergency planning, effects of radiation, and a Limerick plant description.

A draft brochure has been prepared and concurred in by the Pennsylvania Bureau of Radiation Protection for annual distribution to all residents in the 10-mile EPZ. The information in the brochure includes a description of how the public will be notified, what actions the public should take and general information about radiation. The applicant projects that the public information brochures will be distributed by about December 1, 1984. The staff, with the assistance of FEMA, will verify that the brochures have been distributed. Information on emergency planning for Limerick, however, has been provided to the public. Newspaper and radio advertisements covering emergency preparedness issues have been run on a regular basis; a bimonthly company newsletter featuring articles on emergency planning has been mailed to all residents in the 10-mile EPZ; a 4-page flyer containing specific information on the July 25, 1984 exercise and the alert and notification system (the sirens were sounded during the exercise) was also mailed to all residents in the 10-mile EPZ prior to the exercise; and company representatives have spoken on emergency planning matters before various public groups in the area.

The Emergency News Center, located at company headquarters, 2301 Market Street, Philadelphia, will be the principal location for the release of news on the developments during an emergency at the Limerick plant, interviews and news briefings with technical experts, and contact with local governments and residents within the ten-mile radius of the plant. Press briefings will be held at least three times daily and news releases will be distributed at least every three hours. More frequent releases and briefings will be held as necessary.

The Emergency News Center has a designated meeting area to handle media representatives and will be equipped for the use of television cameras, amplifiers and telecommunication equipment. Other sections of the Emergency News Center are designated for interview rooms and for office space for information officers of the Nuclear Regulatory Commission, other government agencies and industry associations. The Manager-Public Information is the supervisor of the Emergency News Center and is the point of contact for release of public information.

Telephone calls from the general public are handled by Customer Service personnel. The Manager-Editorial Services is responsible for insuring that accurate and up-to-date information is made available to the Customer Service people to insure that rumors are countered by accurate information.

Finding

The staff finds that the Applicants' emergency plan meets this Planning Standard, the requirements of 10 CFR 50 Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.8 Emergency Facilities and Equipment

Planning Standard

Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

Emergency Plan Evaluation

Technical Support Center

The Technical Support Center (TSC) is located in the TSC building on the Unit 2 side of the plant, nortn of the administration building. The TSC has approximately 2500 square feet of space and is provided with the same radiological habitability as the Control Room.

The TSC equipment includes:

a) Emergency Response Facility Data System.
b) Radiation and Meteorological Monitoring System.
c) Maps, overlays/nomographs, and calculational aids used in projecting and evaluating offsite doses and in tracking effluents.
d) Supplies and equipment for monitoring teams and other emergency personnel.

e) Sanitary and food preparation facilities.

f) Communication equipment for contact with emergency centers and organizations.

A copy of the following documents are stored in, or adjacent to, the Technical Support Center:

1) General Arrangement Drawings
2) Piping and Instrumentation Diagrams (P&IDs)
3) Electrical schematics
4) Selected piping system isometrics
5) FSAR and Technical Specifications
6) Emergency Plan and Emergency Plan Procedures
7) Plant Procedures

The TSC is activated under Alert, Site Area Emergency and General Emergency conditions.

Emergency Operations Facility

The Emergency Operations Facility (EOF) is located at Philadelphia Electric Company's Plymouth Service Building (Ridge Pike and Chemical Road, Plymouth Meeting, Pa.) which is approximately 17 miles from the plant. The EOF serves as the central location for coordinating response activities between onsite and off-site groups and for the coordination of radiological and environmental assessment. Space and facilities are provided for PECo, NRC, Commonwealth, and other appropriate emergency personnel. The EOF is activated at a Site Area or General Emergency and, when activated, is the central point for the receipt and analysis of all field monitoring data.

The Emergency Operations Facility equipment includes:

a) Communications equipment for contact with emergency centers and organizations.
b) Maps, overlays/nomographs, and calculational aids used in projecting and evaluating offsite doses and in tracking effluents.
c) Supplies and equipment for offsite monitoring teams and other emergency personnel.
d) Sanitary and food preparation facilities.
e) Emergency Response Facility Data System.
f) Radiation and Meteorological Monitoring System.

Operational Support Center

The Operational Support Center is an enclosed space at the 269 foot level of the turbine building immediately outside the primary access doors of the Control Room. It is equipped with dedicated phones to the Control Room, the TSC and the EOF, a Gaitronics plant paging unit, portable radiation monitoring equipment, rapid deployment kits for inplant monitoring teams, air pacs, protective clothing and flashlights. The Emergency Director may direct the Operational Support Center to an auxilary location if determined necessary. The OSC is activated during an Alert, Site Area of General Emergency.

Section 7.3 and Table 7-3 of the LGS Plan identify onsite process and effluent radiation monitoring systems used to initiate emergency measurements. The equipment includes meteorological, and seismic instrumentation. Appendix E lists equipment to be stored at the emergency response facilities, and Section 8.3 states the frequency of inventory checks and calibration of equipment. As discussed in Section 2.3.3 of the SSER, the NRC staff has considered the proximity of the cooling towers to Met Tower 1, and has determined that the data from Met Tower 1 can be relied upon in an emergency.

The Emergency Response Facilities were all activated and utilized during the full-scale exercise of July 25, 1984.

Finding

The staff finds that the applicants' emergency response facilities and equipment are adequate to meet the requirements of 10 CFR 50.47 and 10 CFR 50 Appendix E on an interim basis. Final staff evaluation of the operational capability of the completed emergency response facilities will be conducted as part of the post-implementation review of emergency response capabilities against the requirements of Supplement 1 to NUREG-0737, as described in NRC Generic Letter No. 82-33, dated December 17, 1982.

13.3.2.9 Accident Assessment

Planning Standard

Adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

Emergency Plan Evaluation

Table 4-2 of the LGS Plan identifies plant system and effluent parameter values characteristic of a spectrum of off-normal conditions and accidents including those which correspond to the example initiating conditions of Appendix 1 of NUREG-0654.

A computerized Radiation and Meteorological Monitoring System (RMMS) is employed to assess the offsite radiological impact of emergencies. The RMMS is a computer-based data acquisition and analysis system which provides the capabilities for making near real-time, site specific estimates of atmospheric transport and diffusion and offsite doses during and following an accidental airborne radioactivity release.

The RMMS accesses near real-time release point data and meteorological data from one of two meteorological towers on the site. The meteorological data and release point data are used with site specific terrain conditions to calculate atmospheric dispersion coefficients (Chi/Qs) for each of the sixteen sectors. In the event meteorological data or release point data is inaccessible, manual data entry is possible for all variables used in determining Chi/Qs and doses.

The RMMS data files and calculational capabilities are available to personnel in the control room, TSC, and EOF through interactive consoles located in these facilities. Communication links are also provided to allow for remote interrogation of meteorological parameters and effluent transport and diffusion by the NRC and the appropriate State emergency response agency. As a back-up to the computer-based RMMS, site specific estimates of atmospheric transport and diffusion and offsite doses during and following an accidental airborne radioactivity release can be obtained by using a manual procedure. The manual procedures use pre-determined atmospheric dispersion coefficients based on the same criteria used in the RMMS system.

Should the effluent radiation monitors be off scale or otherwise inoperable, assessment of releases and offsite exposures can be made using the RMMS even though the communication link to the effluent radiation monitor is lost. The RMMS will prompt the operator for manual data entry of necessary release point data. These data can be obtained from containment monitor readings or grab samples.

The percentage of fuel inventory released can be correlated to radioactivity (curies) available for release based on an isotopic spectrum determined by the Post Accident Sampling System (PASS). Procedures and figures are provided to correlate the containment high range radiation monitor readings (R/hr) to the percent of fuel inventory released to the containment atmosphere as a function of time after plant shutdown.

Emergency kits contain radiation survey equipment which enables the Survey Teams to obtain dose rates, surface contamination and airborne radioactivity levels including radioiodine measurements to supplement calculations based on effluent data. These emergency kits are located at emergency facilities outside the plant for ready accessibility. The equipment in these kits is dedicated for emergency use. The applicant will use silver zeolite cartridges in air sampling equipment to detect and measure radioiodine concentrations in air in the presence of a noble gas background.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50, Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.10 Protective Response

Planning Standard

A range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

Emergency Plan Evaluation

Shift supervision (Interim Emergency Director) will notify plant personnel of the existence of an emergency condition which may require initiation of protective actions. The plant public address system and evacuation siren and the river warning system broadcast speakers are capable of providing a warning to employees, visitors, contractors and construction personnel, and other persons who may be in the public access area or passing through the site. It is anticipated that only several minutes should elapse between receiving the alarm and completion of notification.

There are two site evacuation routes at approximately 90 degrees to each other. Table 7-1 of the LGS Plan lists the evacuation assembly areas onsite. The offsite assembly area is at the Limerick Airport north of the plan with an alternate assembly area at the Cromby Generating Station approximately 10 miles east of the site. Evacuees from the site will be monitored at the assembly areas which will have provisions for decontamination of personnel and vehicles. Evacuees will be expected to use their personal vehicles in evacuating to the designated assembly area. Plant access roads are maintained clear in the winter months.

A computer assisted accountability system is provided to afford station security personnel with a method for determination of station personnel accountability with the goal that accountability can be established within thirty minutes. During the July 25, 1984, exercise, the computer system was not fully operational; however, initial accountability was accomplished within 23 minutes. There were delays in locating some of the persons who were not accounted for, which the applicant recognized in its exercise critique and is revising procedures to correct.

For individuals remaining or arriving onsite during the emergency, respiratory equipment is maintained at storage areas and emergency control centers. Self-contained breathing apparatus is provided for Control Room personnel and also contained in the TSC and OSC emergency kits. Supplies of protective clothing including coveralls, rubber gloves, shoe covers and boots, caps and hoods and plastic suits are maintained for normal plant use by health physics personnel and are available in the Control Room, TSC and OSC.

A supply of potassium iodide (KI) tablets is available in the Technical Support Center. The Philadelphia Electric Company Medical Director has authorized the use of KI tablets for emergency workers. The Personnel Safety Team leader is responsible for distribution per approved procedure to specific emergency workers judged in need of thyroid blocking.

Section 5.2.1.1 of the LGS Plan shows that one of the duties of the Interim Emergency Director (Shift Superintendent, or alternate Shift Supervisor) is to provide recommendations for protective actions directly to Pennsylvania officials, or if warranted in a General Emergency, to county officials. Protective action recommendations will be determined in accordance with EP-317, "Determination of Protective Action Recommendations." This procedure directs that recommendations should be made on the basis of plant and core conditions before there is a release of radioactivity from the plant.

The applicant has submitted the document "Evacuation Time Estimates for the Limerick Generating Station" dated May 1984. The report has been evaluated for consistency with the guidance in NUREG-0654, Appendix 4, by the NRC staff and determined to be adequate.

Finding

The staff finds that the Applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50 Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.11 Radiological Exposure Control

Planning Standard

Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

Emergency Plan Evaluation:

Emergency exposure guidelines that conform to EPA guidelines (EPA 520/1-75/001) are provided in Appendices to EP-261, Damage Repair Group; EP-252, Search and Rescue/First Aid; EP-251, Plant Survey Group; EP-250, Personnel Safety Team Activation; EP-230, Chemistry Sampling and Analysis Team Activation; EP-222, Field Survey Group; and EP-220, Radiation Protection Team Activation. The emergency exposure guidelines direct that overexposure in an emergency is to be authorized by the Emergency Director for specific functions such as search and rescue activities or life-saving. These procedures are utilized during emergency preparedness training for all personnel, and it is through such training that the impact of the guidelines will be presented to plant staff. EP-221 directs the issuance of emergency dosimetry, respiratory protection equipment and bioassay, and references health physics procedures for 24-hours per day capability for processing of dosimteric devices. Table 8-1 of the LGS Plan, "Initial Training and Periodic Retraining," is consistent with health physics procedures and training, and includes the need for reading dosimeters at appropriate frequencies and for the maintenance of dose records for emergency workers involved in a nuclear accident.

EP-254, Vehicle and Evacuee Control Group, and EP-255, Vehicle Decontamination, provide guidelines for decontamination of evacuees and vehicles and personnel monitoring. These procedures will be accomplished in accordance with the applicant's health physics procedures for egress frisking of individuals, assembly area monitoring, decontamination, and the forms to be used to record potential data. Onsite personnel decontamination facilities for emergency conditions include showers and sinks which drain to the liquid radioactive waste processing system and cleaning agents are maintained at the primary health physics decontamination area in the plant.

EP-401 lists the special and unique features of radiation protection during an emergency. In the event of radioactive contamination of ground surfaces within the LGS, access to such areas shall be controlled. Should contamination of site drinking water sources be suspected, water samples shall be analyzed, and quarantine established if necessary. Criteria for returning areas and items to normal use are specified in health physics procedures. Appendix E to the LGS Plan lists equipment and supplies available for use in an emergency and its location.

Finding

The staff finds that the Applicant's Emergency Plan meets this Planning Standard, the requirements of 10 CFR 50, Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.12 Medical and Public Health Support

Planning Standard

Arrangements are made for medical services for contaminated injured individuals.

Emergency Plan Evaluation

The LGS Plan includes a letter of agreement with the Pottstown Memorial Medical Center which states that the hospital agrees to accept casualties from a radiation or non-radiation accident at the plant. The hospital has equipment for patient acceptance, emergency surgery, personnel dosimetry and personnel decontamination. Hospital personnel will perform emergency treatment of contaminated patients, including resuscitation and stabilization. If required, more definitive evaluation and treatment would be performed by the Radiation Management Corporation (RMC) which has a staff experienced in nuclear medicine and accident management.

RMC has available, on a 24-hour per day basis, a Radiation Emergency Medical Team (REM Team). The REM Team consists of experienced physicians, health physicists and technicians and has portable medical and health physics equipment to render emergency treatment at accident sites and to conduct the initial evaluation of the radiation status of both patients and the environment. Transportation of the REM Team and its equipment will normally be by truck, but if required can take place by a helicopter converted for use as an ambulance for two litter patients. In regard to onsite medical assistance, the REM Team capabilities include:

1) Consultation and actual assistance to site first aid personnel and the attending physician.
2) Assistance in personnel decontamination.
3) Patient evacuation to Pottstown Memorial Medical Center or to the Radiation Medicine Center of the Hospital of the University of Pa.

RMC personnel have conducted a training program for emergency room physicians and technicians of the Pottstown Memorial Medical Center.

The LGS Plan contains a letter of agreement with Goodwill Ambulance Service to transport accident victims to offsite medical facilities.

The FEMA report, dated May 8, 1984, entitled Interim Finding on the Offsite Radiological Emergency Response Plans for the Limerick Generating Station, states that this planning standard is essentially complete and includes the following information:

> Ambulance services located within, or serving, the plume exposure pathway EPZ will not routinely be used for evacuation support to health care facilities. They would be available for the continued emergency medical service coverage of their service area, including transporting victims of radiological accidents to medical support facilities. Ambulance services located outside and not serving the plume EPZ, and support County ambulance services, will evacuate health care facilities located within the EPZ, evacuate homebound invalids and provide any other needed assistance.

Each operating shift at LGS will have at least one person trained in first aid procedures in accordance with the guidance of American Red Cross. First aid kits are strategically located throughout the site. An onsite medical facility has been established adjacent to the Personnel Processing Center.

The July 25, 1984 exercise included a contaminated, injured individual transported from the plant to Pottstown Memorial Medical Center for treatment. NRC Inspection Report No. 50-352/84-41, dated September 5, 1984, states, "Medical personnel at the hospital demonstrated efficient handling of an injured, contaminated individual (e.g. contamination isolation, dose reduction practices)."

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50, Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.13 Recovery and Reentry Planning and Postaccident Operations

Planning Standard

General plans for recovery and reentry are developed.

Emergency Plan Evaluation

Criteria have been established for de-escalation from an emergency phase to a recovery phase. These include consultation with local, State and Federal officials that station conditions warrant de-escalation. EP-410, Recovery Phase Implementation, specifies key positions in the recovery organization and describes how response organizations are informed.

Total population exposure can be calculated by a routine of the Radiation and Meteorological Monitoring System . Population data in a particular area is fed into the computer and multiplied by the integrated radiation exposure in that area. EP-316 describes the applicants procedure for a manual calculation of total population exposure.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50, Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.14 Exercises and Drills

Planning Standard

Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

Emergency Plan Evaluation

The applicant's emergency plan provides for the conduct of periodic exercises and drills to develop and maintain emergency response skills among the various groups of emergency workers. Annual exercises will be conducted according to the guidance set forth in NRC and FEMA rules to test the integrated capabilities and a major portion of the basic elements within the plan. Offsite organizations as well as the applicant's response organizations will be involved. The scenarios used for the various exercises will contain the essential elements set forth in NUREG-0654 and will be designed to allow flexibility in decision making. At the conclusion of each exercise, a critique will be held as soon as possible. Organizational means for evaluating the results of the post-exercise critique and implementing corrective action are described in Section 8.12 of the LGS Plan.

In addition to the exercises, drills will be conducted covering communications, fires, medical emergencies, health physics, and radiological monitoring. Drills will be supervised instruction periods aimed at testing, developing, and maintaining skills in emergency response task areas. Management control will be established so that necessary corrective actions are implemented.

Each drill and exercise will be conducted to test the state of emergency preparedness and will be designed to meet a list of specific objectives which are specified in the Emergency Plan. The Emergency Planning Coordinator will coordinate and implement revisions to the emergency plan and required corrective actions resulting from the drills and exercises.

On July 25, 1984, a full participation exervise involving both onsite and offsite response was conducted at the Limerick site. Onsite preparedness was evaluated by the NRC while offsite preparedness was evaluated by FEMA. The NRC findings are contained in Inspection Report No. 50-352-84/41 dated September 5, 1984. The FEMA exercise findings are contained in a FEMA report dated September 25, 1984. FEMA identified five significant deficiencies which require resolution prior to exceeding 5% of rated power.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50 Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.15 Radiological Emergency Response Training

Planning Standard

Radiological emergency response training is provided to those who may be called on to assist in an emergency.

Emergency Plan Evaluation:

The Emergency Plan provides for training and qualifying personnel on the emergency tasks for which they are responsible as specified in the Plan. Selected personnel will be trained to assume specific positions in the emergency organization. Actions performed by emergency organization personnel will parallel the individual's routine responsibilities as much as practicable. Annual training will be provided that will effectively ensure that each member of the emergency organization can perform non-routine duties with proficiency. All station non-essential personnel (nonassigned) will receive annual instruction concerning their expected response action during an emergency.

Table 8.1 of the LGS Emergency Plan lists the position or function of each individual in the emergency organization and the specific initial training and periodic training intended for that individual. Table 8.1 also lists the training programs which will be established for emergency support groups. All response groups which are required to report to the Station in order to complete their emergency role will be trained in Station access procedures and organizational control (i.e., the identify of on-site individual(s) responsible for controlling their emergency response activity). Each support group will be instructed as to the Station's capabilities associated with their specific emergency function. In addition to the training specified in Table 8.1, local medical support personnel will participate in an annual medical drill with LGS emergency response personnel.

First-aid training will include courses equivalent to the American Red Cross Multi-Media standard first aid instructional system.

Finding

This staff finds that the applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50, Appendix E, and the guidance criteria of NUREG-0654.

13.3.2.16 Responsibility for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plans

Planning Standard

Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

Emergency Plan Evaluation

The overall authority and responsibility for radiological emergency response planning rests with the Office of the Vice President-Electric Production Department. The Director, Emergency Preparedness Section, Nuclear Generation Division, is the PECO Emergency Planning Coordinator and has overall coordination responsibility for development and updating of the Emergency Plan and for coordinating PECO plans with other response organizations. The LGS Site Emergency Preparedness Coordinator is responsible for maintaining emergency preparedness and verifying that emergency preparedness activities are performed correctly, and for review of emergency preparedness deficiencies identified through drills and exercises. Training for individuals responsible for the emergency planning effort is listed in Table 8-1 of LGS Plan.

The LGS Plan contains a specific table of contents and a cross-reference to the criteria of NUREG-0654. Appendix D contains a list of implementing procedures for the Plan. The Plan includes provisions for distribution of the Plan and approved changes, and for quarterly updating of telephone numbers. Section 8.2.1 describes an annual review of the Emergency Plan by a member of the Electric Production Department staff, not immediately responsible for emergency preparedness, who is appointed by the Superintendent, Nuclear Generation Division. The results of the review will be transmitted to the NRC and the Pennsylvania Emergency Management Agency.

Section 8.6 of LGS EP states that an audit will be performed every two years under the cognizance of the Operations and Safety Review Committee. The audit will inlude the Emergency Plan, implementing procedures and practices, training, testing and interfaces with offsite agencies. The results, findings and recommendations of the auditors shall be documented and reported to the Operations and Safety Review Committee and the Station Superintendent, and records will be retained for five years.

Finding

The staff finds that the Applicant's emergency plan meets this Planning Standard, the requirements of 10 CFR 50, Appendix E, and the guidance criteria of NUREG-0654.

13.3.3 Conclusions

As noted in Section 13.3.1 of this report, during the period of June 11-22, 1984, an onsite appraisal was made of the applicant's capability to implement the emergency plan. As a result of the appraisal (NRC report No. 50-352/84-18 dated August 14, 1984), the applicant made commitments as described in letters to the NRC dated September 7 and 27, 1984. Followup inspections will be made by the NRC to verify the applicant's achievement of these commitments as well as those in the letter to the NRC dated September 18, 1984, and also that a viable siren alert system is installed and operational. On July 25, 1984, a full scale exercise was held of onsite and offsite emergency preparedness. The report of the onsite portion of this exercise is contained in NRC report No. 50-352/84-41 dated September 5, 1984. The NRC staff took both the appraisal and exercise findings into account in reaching the following conclusion on the state of onsite emergency preparedness for Limerick.

Based on the NRC review of the Limerick Generating Station Emergency Plan against the criteria in "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654/FEMA-REP-1, Revision 1, November 1980, and upon the applicant's commitments in letters to the NRC dated September 7, 18 and 27, 1984, the staff concludes that the Limerick Generating Station Emergency Plan provides an adequate planning basis for an acceptable state of emergency preparedness and will meet the requirements of 10 CFR 50 and Appendix E thereto applicable to fuel load and low power operations; i.e., up to 5% of rated power.

After receiving supplemental interim findings and determinations made by FEMA on State and local emergency response plans, and upon satisfactory resolution of the significant deficiencies identified by FEMA during the July 25, 1984 exercise, a supplement to this report will provide the staff's overall conclusion on the status of emergency preparedness for the Limerick Generating Station and related emergency planning zones pursuant to power ascension above 5% of rated power.

14 INITIAL TEST PROGRAM

Preoperational Test Deferrals

By letter J.S. Kemper to A. Schwencer dated July 17, 1984, Philadelphia Electric Company requested deferral of twenty-one preoperational tests for Limerick Generating Station, Unit 1 until after fuel loading. Since that time, in a letter dated October 4, 1984, we have been notified that three of these preoperational tests (1P16.1, 1P79.2B, and 1P79.2C) have been completed. Based on our review, as discussed below, we have concluded that the deferral of the remaining eighteen tests is acceptable.

The requested test deferrals may be grouped in the following three categories: (1) after fuel loading, but completed prior to initial criticality, (2) after fuel loading, but completed prior to opening the main steam isolation valves (MSIV), and (3) after fuel loading, but completed prior to exceeding five percent power. A discussion of these deferrals follows.

Preoperational Tests to be Completed Prior to Initial Criticality

The following preoperational tests would be deferred until after fuel loading, but with completion prior to initial criticality.

1P13.5	Fire Protection Halon System*
1P34.1	Reactor Enclosure HVAC
1P45.1	Feedwater System
1P68.1	Solid Radwaste System (Packaging)
1P68.1B	Radwaste Crane
1P70.1	Standby Gas Treatment System
1P73.1	Containment Atmospheric Control System
1P79.2A	Digital Process Radiation Monitoring System
1P79.2F	Gaseous Effluent Radiation Monitoring
1P83.1	Main Steam System
1P83.3	Steam Leak Detection

We have evaluated the deferral of the above tests for Limerick Unit 1 and conclude that they may be safely deferred as proposed by the applicant. During fuel loading and precritical testing there is no significant source of radioactivity or radioactivity decay heat. Therefore, none of the systems or equipment to be tested by these tests: (1) will be used for maintaining the reactor in a cold, shutdown condition, (2) will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications, (3) are required engineered safety features or will be relied on to support or ensure the operations of required engineered safety features, (4) are assumed to function or credit for functioning is taken in the accident analyses of the facility, as described in the FSAR,

*A roving fire watch will be established until testing is complete.

and (5) are required to process, store, or limit the release of radioactive materials. The bases for judging these deferrals are consistent with bases for judging deferrals on other plant test programs.

Preoperational Tests to be Completed Prior to Opening MSIVs

The following preoperational tests would be deferred until after fuel loading, but with completion prior to opening the MSIVs.

1P33.1	Turbine Enclosure HVAC System
1P43.1	Condenser and Air Removal System
1P72.1	Gaseous Radwaste Recombiners and Filters
1P93.2	Main Turbine Control (EHC) System

We have evaluated the deferral of the above tests and conclude that they also can be safely deferred as proposed. This conclusion is based on the determination that with the exception of 1P79.2C, Main Steam Line Radiation Monitoring and possibly, 1P72.1, Gaseous Radwaste Recombiners and Filters, the systems and equipment involved perform no safety related function. With regard to 1P79.2C and 1P72.1, neither steam nor radioactivity will be present in the portions of the plant serviced by these systems prior to opening the MSIVs. Therefore, these systems will not be needed to support operation until the MSIVs are opened and preoperational test performance can be safely deferred until prior to MSIV opening.

Preoperational Tests to be Completed Prior to Exceeding 5% Power

The following preoperational tests would be deferred until after fuel loading but with completion prior to exceeding five percent of rated power.

1P31.1A,B,C,D	Process Computer System
1P76.2	Post-Accident Sampling System
1P58.2	Redundant Reactivity Control System

The plans for completing and testing the Process Computer System have been further modified by the applicant's letter of September 10, 1984. As stated in the following discussion of construction deferral, the staff finds those proposals acceptable.

The Post-Accident Sampling System (PASS) is not required to shut down or remove decay heat from the reactor or control radioactivity. As stated in further detail in Section 9.3.2 of Supplement 2, PASS is to sample reactor coolant to determine if the reactor core has been damaged. No significant amount of radioactivity or decay heat is generated from testing below five percent power, therefore, the likelihood of releasing significant amounts is very low. Therefore, testing of PASS may be reasonably deferred and completed before exceeding five percent power.

The Redundant Reactivity Control System is a system designed to mitigate an anticipated transient without scram (ATWS). The staff concluded in the SER, NUREG-0991, Chapter 15, that ATWS mitigation on an interim bases (until the Commission's decision on final resolution of the ATWS issue was released) could be adequately handled on a procedural basis. Since the applicant committed to generate emergency procedures based on the BWR Owners Group Guidelines, the staff concluded that the ATWS issue was "resolved for the purpose of issuing a full power license." On the basis of the staff's SER conclusion and as stated later in this section of this report, testing of the Redundant Reactivity Control System may be performed prior to exceeding five percent of rated power.

Based on the preceding considerations, we have concluded that deferral of the preoperational tests for the Limerick Generating Station, Unit 1 as requested by the applicant is acceptable provided appropriate changes are made to the technical specifications where affected by these test deferrals. These changes to the initial test program should be documented by confirmatory FSAR Amendment modifying Section 14.2.4 and the individual test abstracts in Table 14.2 noting these changes apply only to Unit 1. These test deferrals will also be made licensing conditions.

Construction Completion Deferrals

By letter dated July 17, 1984, the applicant requested deferral of six construction completion items. Since that time, by letter dated October 4, 1984, we were notified that the item related to the wrapping of the cables for raceway separation criteria compliance will be completed prior to fuel load. Since this deferral is therefore no longer required, it is not evaluated. The remaining five deferral requests are evaluated below.

1. The applicant proposed to defer completing construction of the redundant reactivity control system until prior to exceeding 5 percent of full power. This deferral is acceptable as stated in Section 7.2.2.5 of this report.

2. The Control Room Design Review Final Report called for enhancements (paint, tape and label) to the control room panels, rescaling some instruments using acceptable human factors methods, and changes to some standard control switch shapes and colors. The applicant proposed to defer the completion of these items until prior to exceeding 5 percent power. These deferrals are acceptable as stated in Section 18.1.7 of this report.

3. The applicant has proposed to defer completion of the post accident sampling system (PASS) until prior to exceeding 5 percent of full power. On the basis of the evaluation of the PASS in Section 9.3.2 of Supplement 2 to the SER and the above discussion on deferral of the preoperational test of the PASS, this deferral is acceptable.

4. The applicant has proposed to defer operability of the Safety Parameter Display System and the Emergency Response Facility Data System until April 1, 1985. On the basis of the evaluation in Section 18.2.9 of this report, this deferral is acceptable.

5. The applicant has proposed to defer demonstration of operability of the process computer system until prior to exceeding 5 percent of full power. The staff's evaluation addresses the portions of the system which monitor reactor operation.

   The applicant's letter of July 17, 1984 states that the digital input-output hardware and software functions of the PCS required for the rodblock circuitry to support and aid in the enforcement of procedural restrictions on control rod manipulations will be operational prior to fuel load. In a supplementary letter dated September 10, 1984, the applicant states that all portions of the computer system which are associated with monitoring reactivity control systems will be tested and verified operational prior to fuel load and that the core performance software portion of the system will be tested during the Power Ascension Program. This supplementary letter also proposed to delete the PCS from the Preoperational Test Program.

The staff questioned the applicant regarding the method of monitoring core power to be used during operations below 5 percent of full power without an operable process monitor. PECo proposes to monitor the core power for Limerick while operating below 5 percent of rated power by the use of the APRM Channels. These are calibrated by performing a uniform heatup rate test at below boiling temperatures. The APRM gains are adjusted to maximum and the core is brought to a low (1-2 percent) power level. Control rods are manipulated to keep the power level constant while the core temperature is increased by 50-100 degrees Fahrenheit. Using the time required for the increase and a value for the heat capacity of the core, a value for core power may be obtained. The gain factors of the APRM Channels are then adjusted so that each channel reads the correct core power. The heat capacity used in the analysis is deliberately made conservative in order to ensure a conservative indicated power. This procedure is routinely performed during startup testing and the results are used as a power indication prior to performing a heat balance at about twenty percent of full power. We conclude that this method of monitoring the core power level is acceptable for use during the time when Limerick is limited to operation below 5 percent of full power.

The staff has determined that on the basis that all portions of the PCS associated with monitoring reactivity control systems will be operational prior to fuel load and the existence of an acceptable method of monitoring core power below 5 percent of rated power, the applicant's proposals for demonstrating operability of the Process Computer System are acceptable.

Initial Plant Test Program Revisions

As a result of amendments to the FSAR submitted since publication of the SER, it was necessary to request additional information (RAI) about changes made to the previously reviewed and approved Initial Plant Test Program. This RAI was transmitted to the applicant by letter dated September 20, 1984. The applicant responded in letters dated September 28 and October 5, 1984. These responses are acceptable as discussed in the following paragraphs. These resolved items are subject to confirmation by FSAR amendment. This review covers through FSAR Amendment 35.

The test abstracts listed below contained inadequate acceptance criteria traceability. In part, this inadequacy resulted from deletion by amendment and part by tests that were added without sufficient description of the source of the acceptance criteria. The test abstracts will be modified to include reference to appropriate FSAR subsections or vendor documentation, which is acceptable to the staff.

1. (P-2.1) 125-V (Div III, IV) dc Safeguard Power System
2. (P-2.2) 125/250-V (Div I, II) dc Safeguard Power System
3. (P-11.1) Service Water System
4. (P-15.1) Turbine Enclosure Cooling Water System
5. (P-53.1) Standby Liquid Control System
6. (P-65.1) Radwaste Enclosure HVAC System
7. (P-70.1) Standby Gas Treatment, Reactor Enclosure Air Recirculation, Secondary Containment Isolation
8. (P-76.2) Post-Accident Sampling System

The concerns listed below about the electrical systems testing also resulted from our review of the recent FSAR revisions.

1. The dc Power System Tests (P-2.1, P-2.2) should reinstate testing of all dc loads necessary for safe shutdown at minimum terminal voltage or provide an acceptable alternate.

2. Deletion of reference to system bus voltages in the acceptance criteria of the 13.2-kV Unit Auxiliary and 4-kV Safeguard Power System test abstracts (P-3.1, P-4.1) had not been justified.

3. There is an inconsistency between the Unit Scope section for the dc Power System tests (P-2.1, P-2.2) and FSAR Subsection 8.3.2.1 pertaining to common or shared dc power systems between Units 1 and 2.

Item 3 was acceptably resolved by deleting the reference to testing of common systems from the Unit Scope of preoperational tests P-2.1 and P-2.2 and making them consistent with FSAR Subsection 8.3.2.1 (i.e., there are no common or shared dc systems).

Item 2 was acceptably justified by noting that (a) voltage regulation of these busses was verified in other preoperational testing and (b) tests P-3.1 and P-4.1 are primarily functional tests of the breaker logic and control circuits, making the voltage values unnecessary.

Item 1 was acceptably resolved by the applicant's commitment to perform a special test to measure voltage at all Class 1E dc distribution busses and at that Class 1E dc equipment which must be preoperational when the battery is at minimum terminal voltage. The results of this special test will be compared to confirmatory analysis to substantiate the analysis' ability to predict the voltage drops between busses and the dc loads. The special test will be performed during the startup test program prior to exceeding 5% reactor power.

In addition to the above matters the staff also noted that in the process of modifying the Loss of Instrument Air test (P-100.2) to conform to Regulatory Guide 1.68.3, an inconsistency resulted relating to conformance of this test abstract and other sections of the FSAR to regulatory guides. This inconsistency was acceptably resolved by deleting the reference to the outdated Regulatory Guide 1.80 and appropriately modifying FSAR Subsection 1.8 to indicate compliance with Regulatory Guide 1.68.3.

Measuring the flow of each MSRV had been deleted from the Main Steam Relief Valves (MSRVs) Performance test (STP-26). This concern was acceptably resolved by reinstating the MSRV flow measurements and appropriate acceptance criteria in the STP-26 Test Abstract.

15 ACCIDENT ANALYSIS

15.6 Decrease In Reactor Coolant Inventory - LOCA

Loss-of-Coolant Accident (Radiological Considerations)

In Section 15.6 of the SER, the staff indicated that the applicant had selected and analyzed a hypothetical design basis loss-of-coolant accident (LOCA) and had shown that the distances to the exclusion area boundary (EAB) and the low population zone boundary (LPZ) in conjunction with the plant's engineered safety features are sufficient to provide reasonable assurance that the radiological consequences of such an accident are within the exposure guidelines of 10 CFR Part 100.11(a)(1) and (2). The analysis has included the following sources and radioactivity transport paths to the atmosphere:

(1) contribution from containment leakage to the reactor building;

(2) contribution from post-LOCA leakage from engineered safety features outside containment; and

(3) contribution from main steam isolation valve leakage.

The staff's review confirms the adequacy of the applicant's containment design concept and site parameters based upon the following:

(1) the applicant's provisions for and design of the containment system, the Standby Gas Treatment System (SGTS), and the Reactor Enclosure Recirculation System (RERS) are acceptable as identified in Chapter 6 of this report; and

(2) The staff's independent analysis of the radiological consequences of a hypothetical design basis LOCA as described below.

STAFF EVALUATION

In a letter dated August 2, 1984, the applicant provided a revised analysis which reflects an increase in the anticipated leakage of the reactor building, a corresponding increase in the secondary containment exhaust rate, and a smaller drawdown period. Therefore, the following revised LOCA analysis is based on the new parameters submitted with the revised analysis.

LOCA - Containment Leakage Contribution

The staff's calculation of the radiological consequences of the hypothetical LOCA used the conservative assumptions of positions C.1.a through C.1.e of Regulatory Guide 1.3 (Revision 2), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors." The primary containment was assumed to leak to the secondary containment at a constant rate of one-half percent of the containment volume per day for the duration of the accident. There was assumed to be no bypass leakage except during those periods when the secondary containment would not be drawn down to at least a -0.25 inch water gauge (wg) pressure. During the periods that the secondary containment would not be drawn down, it was assumed that the primary containment leakage went directly to the environment without credit for mixing or treatment of any kind.

The pressure within the reactor building is maintained at a -0.25 inch wg below atmospheric during normal operation by exhausting the reactor building air through the normal ventilation system. Upon receipt of a safety features actuation signal, the normal ventilation system is to be automatically switched off and the SGTS is actuated. The applicant's analysis indicated that during the changeover, a pressure transient would occur within the reactor building, such that the pressure increases to a slightly positive pressure for a short time and then returns to a negative pressure of -0.25 inch wg at about 135 seconds. At three minutes into the accident, the RERS would be actuated.

The staff evaluated the specific features of the reactor building, the SGTS, and the RERS and noted that the air volume of the reactor building was about four times larger than that of the primary containment (1.8×10^6 cubic feet versus 4.1×10^5 cubic feet). Because the RERS produces high recirculation in the reactor enclosure building, the staff would expect that the primary containment leakage would be thoroughly mixed with the reactor building air prior to treatment by the SGTS. Nonetheless, in the staff's analysis, the primary containment leakage was conservatively assumed to be mixed with only 50 percent of the reactor building air during this period. The assumptions used in calculating the design basis LOCA doses are summarized in the revised Table 15.5 included in this report. The calculated doses resulting from the LOCA are summarized in the revised Table 15.1 included in this report.

LOCA-Main Steam Isolation Valve Leakage Contribution

In addition to the direct leakage from the containment, the LOCA can also lead to activity releases through the main steam isolation valves (MSIVs). Each of the steam lines is equipped with two MSIVs which are closed by a LOCA-generated signal. In addition, for each steam line, the MSIV leakage control system (LCS) collects any leakage from the valves and this leakage is processed by the RERS and SGTS before venting through the plant stack. The MSIVLCS consists of an outboard and an inboard system. The outboard system collects any leakage between the MSIV outside the containment and the turbine stop valve, while the inboard system collects any leakage between the MSIVs located inside and outside containment. The staff has reviewed the MSIVLCS for conformance with Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Nuclear Power Plants" (Rev. 1) and the findings are reported in Section 6.7 of the SER.

In the calculation of the contribution to the LOCA dose, the staff assumed that one of the inboard isolation MSIVs failed to close, thus allowing contaminated steam to travel to the outboard valve, and that the outboard valve would leak at the technical specification leakage limit of 11.5 standard cubic feet per hour (scfh). This leakage would normally be assumed released directly to the environment. However, the Limerick plant is designed with Seismic Category I main steam line piping to the turbine stop valves and the turbine stop valve is seismically supported. Therefore, it is expected that the turbine stop valve would be closed and the outboard system will be actuated before any contaminated steam reaches the turbine stop valve. Nonetheless, it was assumed that the failed MSIV leaked into the steam tunnel and fission products were released to the secondary containment and then to environment by the RERS and SGTS. All of the collected leakage is to be directed to that part of the steam tunnel that is within the reactor enclosure building (and is part of the secondary containment) where it is to be processed by the RERS and SGTS before being released to the environment.

The MSIVLCS was assumed to be actuated 20 minutes into the accident and functions for the duration of the accident. In addition, each of the four main steam lines was assumed to leak at the technical specification leakage limit of 11.5 scfh (total leakage is 46 scfh).

The calculated doses resulting from this release path are given in the revised Table 15.1 included in this report.

LOCA-Leakage From Engineered Safety Feature Systems Outside Containment Contribution

Leakage from engineered safety features (ESF) components outside the primary containment also would release iodines to the secondary containment, then the iodines would be mixed within the secondary containment with activity from the primary containment leakage. Releases to the environment are to be treated by the RERS and SGTS. The applicant has indicated that during the postulated post-accident operation the normal leakage from engineered safety feature components outside the primary containment will be small. However, the applicant assumed a 5 gallon per minute (gpm) leak for the analysis; the staff also used this leak rate because the staff considered it conservative. The results of the staff's calculations are summarized in the revised Table 15.1. Because the applicant has provided an engineered safety feature grade filtration system which will filter the reactor enclosure building exhaust, the staff has not calculated the contribution to the LOCA doses resulting from a passive failure in an ESF component (as specified in SRP Section 15.6.5, Appendix B).

Staff Findings

The staff has reviewed the applicant's analysis and has performed an independent analysis of the radiological consequences from each of these transport paths. The staff's assumptions are presented in the revised Table 15.5 of this report. The calculated thyroid and whole body doses from the hypothetical LOCA are listed in the revised Table 15.1 of this report.

The staff concludes that the distances to the EAB and LPZ of the Limerick site, in conjunction with the engineered safety features of the Limerick plant, are sufficient to provide reasonable assurance that the total radiological consequences of such an accident will be within the exposure guidelines set forth in 10 CFR Part 100, Paragraph 11. This conclusion is based on the staff review of the applicant's analysis and on the results of the independent analysis performed by the staff which confirms that the calculated doses are within these guidelines.

15.7 Radioactive Releases From a Subsystem or Component

Fuel Handling Accident

In Section 15.7 of the SER, we evaluated a fuel handling accident using assumptions consistent with Positions C.1.a through C.1.k of Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." The kinetic energy of a single falling fuel assembly was assumed to be perfectly transmitted to the impacted fuel assemblies, breaking open the maximum possible number of fuel rods. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods was assumed to occur, followed by the release of these fission products through the pool water. Radiation monitors located within the normal ventilation system have been installed to provide signals to initiate shutdown of the normal ventilation system and to activate the SGTS. The design is such that the system automatically responds to a radioactivity release from the pool as a result of a fuel handling accident, either within the containment or the spent fuel pool area, and no significant fraction of the fission product release escapes untreated (the SGTS is described in Section 6.5 of the SER). However, in a letter dated August 14, 1984, the applicant submitted a revised analysis which reflects an increase in the number of fuel rods that could be damaged. Therefore, the assessment of the fuel handling accident has been revised based on the new parameters submitted with the revised analysis.

A list of the assumptions obtained for Limerick based upon Regulatory Guide 1.25 positions is given in the revised Table 15.4 in this report. The offsite doses computed using these assumptions are listed in the revised Table 15.1, and are well within the guideline dose limits of 10 CFR Part 100. The staff, therefore, concludes that the plant features designed to mitigate the consequences of fuel handling accidents are adequate.

The above analysis and conclusions are based on the assumption that the Standby Gas Treatment System (SGTS) is operational on the refueling floor. The applicant will not have the SGTS operational until prior to the first refueling outage. The acceptability of this plan is discussed in Section 6.2.3 of Supplement 2 to the SER.

Spent Fuel Cask Drop Accident

In Section 9.1.5 of the SER, the NRC staff concluded that the overhead heavy load handling systems were adequately designed to prevent a cask drop accident causing radioactive releases from the spent fuel in excess of that assumed in the fuel handling accident. Based upon this finding, and in compliance with SRP 15.7.5, no radiological consequences of a cask drop accident have been computed.

15.9 TMI Action Plan Requirements

15.9.3 II.K.1, IE Bulletins on Measures to Mitigate Small-Break LOCAs and Loss-of-Feedwater Accidents

II.K.1.5 Assurance of Proper Engineered Safety Features Functioning

In the SER the staff noted as a confirmatory item that it would verify that administrative procedures addressing valve positioning requirements, positive controls and test and maintenance activities associated with engineered safety features satisfied the requirements of IE Bulletin 79-08, Item 6.

The NRC Region I office conducted inspections of the applicant's response to IE Bulletin 79-08 in regards to item II.K.1.5 and reported the results in Inspection Report 50-352/84-36; 50-353/84-10 as transmitted by letter to the applicant on August 20, 1984. The portion of the report addressing this issue is as follows:

> (Closed) II.K.1 IE Bulletins and Measures to Mitigate SBLOCA's and loss of FW Accident (Items 5, 17, 20, 21, 22 and 23)
>
> This item grouped IE Bulletins 79-05, 79-05A, 79-06A, 79-06B and 79-08 together, which were each issued as a result of the Three Mile Island Unit 2 incident. Each of these bulletins was issued to the licensee for informational purposes only. The inspector verified that the licensee received each bulletin and conducted an adequate review, taking appropriate action where necessary. Inspection Report 50-352/81-17 documents the closure of some of these bulletins and bulletin 79-08 is closed in paragraph 4 of this report. Items 5, 22 and 23 have been adequately addressed by the licensee while items 17, 20, and 21 were not applicable to the licensee's plant type.

Conclusions

On the basis discussed above, the staff concludes that this issue is closed.

15.9.5 III.D.1.1 Primary Coolant Outside Containment

In the FSAR, the applicant provided a summary description of their program to reduce leakage from systems outside containment that could contain highly radioactive fluids following a transient or accident to as-low-as-practical levels. This description included the systems to be leak tested and the testing methods to be employed. The staff concluded in the SER that, based on its review of this information, the design of the leak test program meets the requirements of Item III.D.1.1. of NUREG-0737 and is therefore acceptable.

In a letter dated August 24, 1984, the applicant provided additional information regarding a program to reduce potential leakage paths due to design and operator deficiencies as required in NUREG-0737. Based on the staff's review, the considerations given to this aspect of the program are acceptable.

In a letter dated September 12, 1984, the applicant provided additional information concerning the submittal of initial leak test results as required in NUREG-0737. The applicant submitted by letter dated October 12, 1984, the results of the following leak reduction program surveillance tests that deal with contaminated pipe inspection tests:

(1) Scram Discharge Volume System,
(2) Residual Heat Removal System,
(3) Core Spray System, and
(4) Safeguard Piping Fill System.

The remaining tests will be performed after fuel load since the systems to which they apply are not required to be in surveillance for the fuel load operating condition. Based on our review, this is acceptable.

The contaminated piping inspection tests for the Post Accident Sampling System will be completed prior to exceeding 5 percent power and the test for the Post LOCA Recombiners will be completed prior to Startup (Technical Specification Operating Condition 2). The tests for the High Pressure Core Injection and Reactor Core Isolation Cooling Systems will be conducted when reactor pressure reaches normal operating pressure. The results of these tests will be submitted after the last test has been performed. Based on our review, this is acceptable.

Based on the above, the applicant's proposed program to reduce leakage from systems outside containment that could contain highly radioactive fluids following a transient or accident to as-low-as-practical levels is acceptable.

Table 15.1 Radiological consequences of design-basis accidents

Postulated accident	Exclusion area[1] 2-hour dose (rem) Thyroid	Exclusion area[1] 2-hour dose (rem) Whole body	Low population zone[2] 8-hour dose (rem)[3] Thyroid	Low population zone[2] 8-hour dose (rem)[3] Whole body
Main steam line failure outside containment				
w/concomitant iodine spike	4.2	1.3	1.0	1.4
w/pre-accident iodine spike	83	1.3	18	1.4
Rod drop accident	0.7	0.1	1.0	0.04
Fuel-handling accident	2.3	0.5	1.5	0.3

LOCA

Duration, hrs From	To	Exclusion Area Thyroid (Rem)	Exclusion Area Whole Body (Rem)	Low Population Zone Thyroid (Rem)	Low Population Zone Whole Body (Rem)
Containment Leakage					
0.0	2.0	107	3	23	1
2.0	8.0	-	-	1	2
8.0	24.0	-	-	1	2
24.0	96.0	-	-	3	2
96.0	720.0	-	-	3	1
ECCS Leakage					
0.0	2.0	35	<1	8	<1
2.0	720.0	-	-	3	<1
MSIV Leakage					
0.0	2.0	14	<1	3	<1
2.0	720.0	-	-	1	1
Total LOCA Doses		156	4	46	10

ATMOSPHERIC DISPERSION VALUES (χ/Q) VALUES USED IN ACCIDENT EVALUATIONS

Time Period	χ/Q VALUE (sec/m^3)
0-2 hour (EAB)[1]	6.4×10^{-4}
0-8 hour (LPZ)[2]	1.4×10^{-4}
8-24 hour (LPZ)[2]	1.1×10^{-4}
1-4 day (LPZ)[2]	5.8×10^{-5}
4-30 day (LPZ)[2]	2.4×10^{-5}

[1]Exclusion Area Boundary (EAB) Distance = 731 meters
[2]Low Population Zone (LPZ) Boundary = 2043 meters
[3]The calculated LPZ doses after 8 hours for the above accidents other than LOCA were determined to be negligible.

Table 15.4 Assumptions used in computing fuel handling accident doses

Reactor power	3458 MW_t
Peaking factor	1.5
Rods failed	212
Total rods in core	47,368
Decay time prior to accident	24 hours

Table 15.5 Assumptions used to evaluate the loss-of-coolant accident

Power Level (Mwt)	3460
Operating Time (years)	3
Core Fraction Airborne in the Drywell (%)	
Noble Gases	100
Iodines	25
Primary Containment Leakrate (% per day)	0.5
Containment Free Volume (ft^3)	4.1×10^5
Reactor Enclosure Free Volume (ft^3)	1.8×10^6
Reactor Enclosure Mixing Fraction (%)	50
Standby Gas Treatment System Flow Rates (ft^3/minutes)	
0-2.5 minutes	2800
2.5 minutes to end of accident (720 hours)	1250
Reactor Enclosure Recirculation System Flow Rate (ft^3/minute)	60,000
Standby Gas Treatment Filter Iodine Efficiencies (%)	
Elemental	99
Organic	99
Particulate	99
Reactor Enclosure Recirculation System Iodine Filter Efficiencies (%)	
Elemental	90
Organic	30
Particulate	99
Minimum Exclusion Area Boundary (EAB) (meters)	731
Low Population Zone Distance (LPZ) (meters)	2043

17 QUALITY ASSURANCE

17.5 Independent Design Verification Program (IDVP)

17.5.1 Background

By letter dated January 10, 1984, the staff requested the applicant to present within 30 days its plans for providing additional assurance that Limerick Unit 1 has been designed and constructed in accordance with the regulations and Safety Analysis Report commitments. As a result of that request, the applicant and the applicant's IDVP contractor, Torrey Pines Technology (TPT), presented their plans in a May 9, 1984 public meeting. At the conclusion of the meeting, the staff verbally approved TPT's Program Plan, subject to resolution of staff comments. Subsequently, the staff documented its approval of and comments on the Program Plan in a May 15, 1984, letter to the applicant. On the same day, May 15, 1984, TPT held a kick-off meeting with the Architect Engineer for Limerick. In a telecon of May 21, 1984, between the applicant, TPT and the staff, the staff's comments were resolved. Rev. A of the Program Plan was provided by the applicant's letter dated June 6, 1984, and formal staff approval was provided in a letter dated July 9, 1984, to the applicant. Per Rev. A of the Program Plan, Torrey Pines Technology's forecast for issuance of the final report was August 31, 1984.

To help provide an early identification and resolution of staff comments, a decision was made by the staff to conduct an IDVP implementation review of TPT. The staff conducted this review on July 24 and 25, which was the earliest date available to perform a substantive review of the technical details being evaluated by TPT. The staff's comments resulting from this implementation review were identified in a exit meeting with TPT on July 25, 1984 and were identified to the applicant in a telecon on July 27, 1984. Potential resolutions of staff comments were discussed in a telecon between the applicant, TPT, QUAB of I&E, and LB-2 of NRR on August 9, 1984. As a result of agreements reached in that telecon, TPT's depth of review was increased (primarily in the civil/structural area) and it was agreed the final report would include the technical review details which substantiate the conclusions of the IDVP. To accommodate these staff concerns and to evaluate an additional 14 potential finding reports, TPT has rescheduled issuance of the final report from August 31, 1984 to late October 1984.

Since the staff's review of the final report and preparation of the associated SSER was not expected to be completed prior to the scheduled fuel loading date, the staff initiated two actions. First, the applicant and TPT were requested, by a letter from the staff dated September 12, 1984, to provide independent assessments concerning whether or not the IDVP work, in progress, had identified any adverse finding that could potentially delay fuel loading and ascension to 5% power. Responses from the applicant dated September 25, 1984 and TDT dated September 21, 1984 both concluded that the low power operating license should not be delayed based upon the IDVP. Second, the staff conducted a review of the IDVP potential finding reports current through September 24, 1984. This review was conducted on September 25 and 26, 1984 at the offices of TPT and the results are discussed in Section 17.5.3.

17.5.2 Results of the IDVP

No results have been documented, since the final report is not scheduled to be issued until late October, 1984.

17.5.3 Assessment by NRC Staff

The staff's implementation review, conducted from July 24 to July 25, 1984, identified no safety concern associated with the design of Limerick Unit 1. Also the staff's review of the IDVP potential finding reports (current through September 24, 1984), did not identify any safety concern which would warrant delay in granting the low power operating license for Limerick Unit 1.

The staff's review of the final IDVP report including the applicant's associated corrective action plans will provide a basis for determining whether the IDVP impacts the granting of a full power operating license.

17.5.4 Conclusion

The IDVP for Limerick Unit 1, as with all IDVPs, is confirmatory in nature. Completion of the IDVP is not a prerequisite to fuel loading. Based upon the staff's IDVP implementation review conducted from July 24 to July 25, 1984; the assessments from the applicant and TPT dated September 25, 1984 and September 21, 1984; and the staff's review of all potential finding reports current through September 24, 1984, the staff finds that the completion of the IDVP should not prohibit granting the low power operating license for Limerick Unit 1.

18 HUMAN FACTORS ENGINEERING

18.1 Detailed Control Room Design Review

All licensees and applicants for an operating license are required to conduct a Detailed Control Room Design Review (DCRDR) in response to NRC Task Action Plan Item I.D.1 (NUREG-0660, May 1980; and NUREG-0737, November 1980 as supplemented by Generic Letter 82-33, December 17, 1982). The purpose of the DCRDR is to identify and correct human engineering discrepancies (HEDs) which might affect the operator's ability to prevent or cope with an accident. NUREG-0700, "Guidelines for Control Room Design Reviews," dated September 1981, provides guidance for conducting the DCRDR.

18.1.1 Background

The staff reviewed Limerick's Program Plan (Reference 1) submitted by the Philadelphia Electric Company (PECO) for the Detailed Control Room Design Review (DCRDR). The staff review of the Program Plan concluded that if the activities described in the Plan are properly executed, they should define and correct the major Human Engineering Deficiencies (HEDs) which exist in the control room. Details on these review results are reported in Reference 2. The staff, with the assistance of Lawrence Livermore National Laboratory (LLNL) conducted an In-Progress Audit of the DCRDR being executed by Limerick. The purpose of the In-Progress Audit was to evaluate: (1) the applicants conformance to the Program Plan, and to the requirements for a DCRDR as stated in NUREG-0737 Supplement 1 and (2) to evaluate review results to date. The In-Progress Audit was conducted during December 6-9, 1983, at the Limerick Station near Pottstown, PA. The results of this audit are defined in a Technical Evaluation Report (TER) (Reference 3) prepared by LLNL.

During the In-Progress Audit, the staff's audit team determined that the Limerick DCRDR did not meet the requirements of Supplement 1 to NUREG-0737 for the following items:

1. The performance of system function and task analyses to determine operator information and control requirements during emergency operations, and,

2. The comparison of display and control requirements which were determined by the function and task analyses with a control room inventory to identify missing displays and controls.

In a recent meeting (Reference 4) with the NRC staff, representatives of the BWROG Emergency Procedure Guidelines (EPG) and Control Room Design Review (CRDR) Committees discussed the task analysis requirements of Supplement 1 to NUREG-0737 (Generic Letter 82-33). The purposes of the meeting were (1) for the Owners' Group to discuss how the EPG development effort and the CRDR program addressed operator information and control needs, and (2) for the staff to determine any additional analyses or documentation needed for review of applicant and licensee submittals on the Detailed Control Room Design Review and Emergency Procedure Generation Package.

The staff concluded that:

1. Based on the presentations by Messrs. Stratman and Migas and the ensuing discussion, it appears that Revision 3 of the EPG provides a functional analysis that identifies, on a high level, generic information and control needs. However, these EPGs do not explicitly identify the plant-specific information and control needs, which are necessary for preparing emergency operating procedures and determining the adequacy of existing instrumentation and controls.

2. Because detailed plant-specific information and control needs cannot be extracted directly from the EPGs, plant-specific analysis is required.

3. Each licensee and applicant must describe the process used to identify plant-specific parameters and other plant-specific information and control capability needs and must describe how the characteristics of needed instruments and controls will be determined. These processes may be described in either the Procedure Generation Packages or the DCRDR Program Plan with appropriate cross-referencing.

4. For each instrument and control used to implement the EOPs, there should be an auditable record that defines the necessary characteristics of the instrument or control and the bases for that determination. The necessary characteristics should be derived from analysis of the information and control needs identified in NRC approved EPGs and from analysis of plant-specific information.

The staff recommended (Reference 3) that Limerick's Systems Function and Task Analysis incorporate the above described activities in the process of completing the DCRDR.

In response to a requirement (NUREG-0737, Supplement 1) to submit a Summary Report of the completed DCRDR, the applicant submitted a Final Report (Reference 5). The purpose of the Summary Report is to describe the results of the DCRDR and to outline proposed control room changes, including their proposed schedules for implementation. The staff was assisted in the review of this report by LLNL.

To report the results of their review of the Final Report, LLNL prepared a Technical Evaluation Report (TER), which was transmitted to the applicant by letter on October 16, 1984. The TER stated that the Limerick Summary Report had been reviewed and many items were found to be inadequate to meet the intent of Supplement 1 to NUREG-0737. Upon analyzing the conclusions presented in the draft TER, the staff determined that the applicant's Final Report had insufficient detail and explanations on the review methods and process to allow for an evaluation of review results.

The staff's review of the Final Report also identified a problem with a major step in the design review. The problem had to to with the completion of task analysis within the DCRDR, wherein the task analysis was reported as incomplete. Based on this problem, and the problems defined by the LLNL evaluation, the staff requested (Reference 6) that a meeting be held with Limerick to discuss these issues.

A meeting was held with PECO on August 7, 1984 in Bethesda, Maryland. The purpose of the meeting was to discuss outstanding items in the detailed control room review which resulted from the evaluation of the applicant's DCRDR Final Report. The minutes of this meeting are recorded in Reference 7.

At the meeting of August 7, 1984, the staff defined and discussed the concerns on the Limerick DCRDR Final Report. PECO responded that details on methods and processes used in the review were not included in the Final Report, but did exist at the Limerick plant site. The NRC staff and LLNL met with PECO on August 8-9, 1984 at the plant site to audit and evaluate the documented method and procedures used by Limerick to conduct the review (minutes of meeting also reported in Reference 7). The TER from LLNL contains the results from the evaluation of Limerick's Final Report and the findings of the audit conducted August 8-9, 1984, at the plant site in Pottstown, Pennsylvania.

This Safety Evaluation Report serves to compile the staff's review results to date. These results are based upon the review of a Program Plan, an In-Progress Audit of the DCRDR, and a review and audit of the Final Report. The Regulatory requirements stated in NUREG-0737 Supplement 1 for a Detailed Control Room Design Review served as the basis for the evaluation which follows.

18.1.2 Evaluation of Detailed Control Room Design Review

The staff's evaluation of Limerick's DCRDR evaluated the review process and sampled review results for compliance with the requirements stated in NUREG-0737, Supplement 1. The main elements of these requirements are: Review Team and Review Program, System Function and Task Analysis, Control Room Inventory, Control Room Survey, Assessment of HEDs, Selection of Design Improvements, Verification That Improvements Will Provide Necessary Corrections and Will Not Introduce New HEDs, and Coordination of Control Room Improvements with other programs. Each of these elements are discussed in the text that follows.

18.1.3 Review Team and Review Program

A qualified multidisciplinary review team and a review program incorporating accepted human engineering principles are required to conduct a control room design review (NUREG-0737, Supplement 1). Guidelines for team selection are found in NUREG-0700 and NUREG-0801.

The staff reviewed the applicant's Program Plan (Ref. 1) for the disciplines, qualifications and experience of the applicant's review team personnel. With the assistance of LLNL, we also reviewed the applicant's Summary Report (Ref. 5). LLNL's TER found the review team to contain all of the disciplines recommended for a DCRDR. We also reviewed the qualification of individual review team members and concluded they were adequate.

The staff also evaluated the Summary Report (Ref. 5) for the operational effectiveness of the review team as a unit. We found that a verification of human engineering suitability of control room panels was performed as an independent review by human factors personnel who were also qualified in nuclear operations. A top-down analysis was conducted for all panels examining functional and spatial arrangement both within each panel and between panels. The analysis used panel arrangement drawings, technical and training material, and instrumentation drawings. The results of this analysis are detailed in the Final Report (Ref. 5) and serve as a firm functional basis for the conduct of the review. Based on this data, the staff concludes a qualified review team has been assembled for the review.

The staff also reviewed the applicant's Program Plan (Ref. 1). Our review concluded an appropriate approach to the DCRDR had been planned and that the applicant had an understanding of the objective of the review and of the review processes. During our In-Progress Audit of the applicant's DCRDR, we were unable to confirm our initial conclusions for all aspects of the review (Ref. 3), and we made recommendations to the applicant to improve upon the review.

LLNL's TER of the applicant's Summary Report initially concluded inadequate information was provided to perform an assessment of the report. A second on-site audit of Limerick's DCRDR was conducted and obtained the necessary information to complete the evaluation of the Summary Report. The results from this audit are also included in the TER.

The staff has evaluated the TER (Appendix A) provided by LLNL and concurs with the general findings and general recommendations made therein. The staff's review results in the form of specific findings and specific recommendations are discussed under the individual review elements which follow. Further, while the DCRDR is incomplete at this time, the applicant is to report on the completed review with supplements to the Final Report.

18.1.4 System Function and Task Analysis

Supplement 1 to NUREG-0737 requires the applicant to perform systems function and task analyses to identify control room operator tasks and information and control requirements during emergency operations. Furthermore, Supplement 1 to NUREG-0737 recommends the use of function and task analyses that had been used as the basis for developing emergency procedures technical guidelines and plant-specific emergency operating procedures to define these requirements.

The background of the applicant's efforts regarding task analysis has been previously discussed in this report (see Background). Also, from the results stated in the TER, we conclude the applicant has not completed the effort required for the task analysis element of the DCRDR.

By letter dated August 16, 1984 (Ref. 8), PECO requested a delay in completion of the task analysis until June 1985, a post license date. The reason given for the delay was that an undocumented task analysis had been previously performed and no Priority 1 (High Safety Significance) HEDs were found for existing instrumentation and controls. Also, in the Final Report (Ref. 5), Limerick did commit to perform a follow-up task analysis on the Emergency Operating Procedures to meet the requirements defined by the NRC, but noted these requirements were defined at a late point in their review process.

Further, during the In-Progress Audit (Ref. 3), the staff did confirm that an adequate task analysis had been performed for additional instruments and controls added to the existing control board, and which were needed by operators to execute the EOPs.

Based on the data and information available, and the discussion in Appendix A, the staff concludes that the completion of the task analysis by June 1985 is acceptable. This position will be made a condition of the license.

18.1.5 Control Room Inventory

Supplement 1 to NUREG-0737 requires that a control room survey be conducted to identify deviations from accepted human factors principles. NUREG-0700 provides guidelines for conducting a control room survey.

LLNL's TER states that as the task analysis has not been completed, it is unlikely that a top-down analysis of sufficient depth and scope was developed and used to determine missing controls/displays. The performance/execution of the task/systems functions analysis, which is specific to Limerick, should generate control requirements needed for the inventory comparisons, which have not been made. The Limerick Summary Report is presently deficient in meeting the requirements of NUREG-0737, dealing with these inventory comparisons. The staff agrees with the conclusion and has addressed this subject within the position established on task analysis.

18.1.6 Control Room Survey

Licensees/applicants are to conduct a control room survey to identify deviations from accepted human factors principles. This survey will include, among other things, an assessment of the control room layout, the usefulness of audible and visual alarm systems, the information recording and recall capability, and the control room environment (NUREG-0737, Supplement 1).

During the In-Progress audit, the staff's audit team reviewed the BWROG survey report. This report was also submitted to the NRC as Appendix B, "BWR Owners Group Control Room Improvements Committee Human Factors Design Review of the Limerick 1 and 2 Control Room, Summary Report," in the Final Report (Ref. 5). The BWROG Summary Report states that the Limerick control room design was found to follow human factors guidelines in many areas; e.g., anthropometric guidelines, functionally grouped controls, etc. However, the BWROG Summary Report identified several significant areas of HEDs. Some of these were:

- Some controls and displays are not inside anthropometric bounds and relocation should be considered,
- Functional grouping of controls and displays could be enhanced with labels and demarcation,
- The Emergency Service Water panel layout is crowded and confusing.

The staff's audit team independently evaluated the above findings and concurred with the results stated by the BWROG.

A number of incomplete survey areas are described in the BWROG Summary Report. These areas include panel layout and design, instrumentation and hardware, annunciators, computers, procedures; and control room environment. The staff's audit team found several additional control room HEDs that were not included in the BWROG Summary Report, which indicated the survey was incomplete.

The applicant stated that the supplemental Control Room Survey was done using checklists developed by the BWROG in order to complete and update the initial survey data. The survey process included panels which were installed after the initial BWROG survey was made. These panels were evaluated against both the initial and supplemental BWROG checklists. Panels which had undergone design changes since the initial surveys were reviewed to determine if the change affected any of the initial HED results. All HEDs from the BWROG Control Room Survey and from the supplementary review were recorded on HED Assessment Forms.

In the Final Report (Ref. 5), the applicant states that several elements in the control room survey could not be completed until control room construction is completed. The areas of the survey which are incomplete were defined as:

- Illumination,
- Atmosphere,
- Noise,
- Verbal Communication,
- Emergency Equipment,
- Computers.

Also, in LLNL's TER, they listed several HED's, identified by the staff's audit team during the In-Progress audit that have not been resolved due to the construction in the control room.

In the Final Report (Ref. 5), the applicant states that the remaining surveys will be completed when appropriate, or will be reviewed and assessed when available, but the applicant does not provide a schedule for the assessment and completion of this work. The staff finds the lack of a schedule unacceptable as HEDs with a high safety significance may result from the surveys which are incomplete.

During a phone conference with PECO on August 16, 1984, the staff raised the issue of a schedule for the completion of the survey. PECO responded that all surveys, with the exception of the human factors evaluation of the computer based SPDS, would be completed and the results presented to the staff in a supplement to the Final Report by October 31, 1984. Based on this data, the staff will condition to the license to ensure completion of the survey and the correction of high safety significance HEDs which may result from the survey.

18.1.7 Assesment of HEDs

Supplement 1 to NUREG-0737 requires that HEDs be assessed to determine which HEDs are significant and should be corrected. NUREG-0700 and NUREG-0801 contain guidelines for the assessment process.

The Final Report (Ref. 5) describes how an HED Assessment form was used to identify, record and manage HED data. The form was used to record each HED by number, to identify criteria used and the source, and to describe the specific discrepancy. All previously defined HEDs, such as from the BWROG survey and operator interviews were converted to these forms. All recently defined HEDs, such as those from the Supplemental Survey, Operator Experience Review, and Licensee Event Report Review were also recorded on the standard assessment forms.

The review team collected all HEDs and sorted them into three categories:

- Those that can be resolved by enhancements,
- Those that form a class of problems that could be part of a common resolution,
- Those that must be considered individually because of their unique nature.

Enhancement design was commenced as the first step of the assessment phase. The design process considered enhancement criteria, and proposed enhancements as developed by consultants and evaluated by the review team. All panel enhancements and new label terminology was reviewed by the team and by additional operating personnel. Revisions were made by the team and the resulting enhancements were placed on the full scale mockup. The Final Report states a large number of HEDs were corrected by enhancement design and are to be implemented in the control room by fuel load.

In a recent letter to the NRC (Ref. 9), PECO proposed that the completion of enhancements (paint, tape, and label) to the control room panels, re-scaling some instruments using acceptable human factors methods, and changes to some standard control switch shapes and colors be deferred from fuel load until prior to exceeding 5 percent power. PECO stated that the deferral will have no impact on the safe operation of Limerick. First, operator training has been conducted on the Limerick simulator which does not as yet incorporate the above human factors enhancements; thus appropriate and timely operator response to an accident would be unaffected by deferral of these enhancements. Second, because of the low level of decay heat present at 5 percent power, significantly more time is available to the operators to consider and initiate mitigative actions than would be available at full power.

The reasons stated by PECO for the delay in the completion of enhancements are acceptable to the staff. However, the staff is concerned that additional delays may result in the completion of these enhancements to post 5 percent power operation. Under these circumstances, the decay heat present will be significantly greater than that present prior to 5 percent power. Furthermore, the HEDs associated with the enhancements have not been assessed for safety significance. Because of the potential exposure to a safety significant HED which remains incomplete upon exceeding 5 percent power, the staff will condition the license to require completion of these prior to exceeding five percent power.

HEDs categorized as class problems, and those to be considered individually, were subjected to an assessment of significance and safety implications and assigned a priority for resolution. A significance checklist was completed for each HED. The significance checklist is defined in Figure 1-2 of the Final Report (Ref. 5). The significance checklist is structured to evaluate human performance in terms of the HED with regard to physical performance, sensory/perceptual performance, mental performance including mental workload. These items are then compiled into a HED significance rating to indicate an overall probability of the discrepancy causing operator error.

LLNL's technical evaluation of the significance checklist initially concluded that the specific checklist is inadequate. Further, they state that the Final Report does not contain the information needed to evaluate the method (significance checklist) and the metric (rating scale, probability of the discrepancy causing operator error). A second on-site audit of the DCRDR was conducted on August 8-9, 1984 to obtain the information needed to complete the evaluation of the assessment method. The staff concluded that minimal compliance with the requirements of HED assessment (NUREG-0737, Supplement 1) is being achieved.

The Final Report states that having decided the significance of an HED, the review team then assessed the safety aspect of the discrepancy. In determining the safety significance of an HED, the combined judgement of the team was used in considering the specific condition caused by the HED or combination of HEDs. The team considered the following factors in their decisions:

- HEDs that cause errors on systems that directly effect safety,
- The potential for violation of technical specification,
- HEDs that are known to have caused errors that will lead to unsafe operation,
- HEDs that could cause the inadvertent activation of a safety related system or a system needed to safely shut down the plant.

Having decided a safety significance, a priority for resolution was determined and assigned. The HED priority and schedule for resolution are stated in the Final Report as:

PRIORITY 1	HIGH SAFETY SIGNIFICANCE (For completion by Fuel Load)
PRIORITY 2	LOW SAFETY SIGNIFICANCE (For completion after Fuel Load)
PRIORITY 3	OPERATIONAL RELIABILITY (For completion after Fuel Load)
PRIORITY 4	NO SIGNIFICANT IMPROVEMENT (HED not corrected)

The methods used to evaluate HED safety significance and priority for resolution appear reasonable and are acceptable to the staff.

The following presents a synopsis of HED distribution as stated in the Final Report:

1.	Number of discrepancies corrected by Fuel Load	88
2.	Number of discrepancies assessed to be acceptable, corrected or no change required	61
3.	Number of discrepancies scheduled for correction subsequent to Fuel Load	36
4.	Held out for further review	4
	TOTAL	189

In LLNL's technical evaluation of the applicant's Final Report, they list 38 HEDs and recommend obtaining clarification on the proposed schedule for resolution. The staff evaluated these HED's and noted that one, HED No. A1-13 had a priority of 1 (high safety significance) but was not scheduled to be fixed until the first refueling outage. In evaluating the remaining HEDs, we noted HED No. SI4-04 also had a priority of 1 but was not scheduled to be fixed until the first refueling outage.

The staff contacted PECO and requested justification for all HEDs assessed with a priority of one but were not scheduled to be fixed until after loading of fuel.

PECO in its response (Ref. 8) stated that in the review of human engineering discrepancies (HEDs), the CRDR team identified four priority 1 HEDs, (high safety significant HEDs). Two of these were corrected (HEDs I5-01 and SI5-03), while the Final Report requested that the correction of the other two (HEDs A1-13 and SI4-04) be deferred to the first refueling outage. Further, they stated that they have re-evaluated the proposed implementation date of HED A1-13 "Annunciator Silence Button." As a result of this re-evaluation, a bell with a softer tone will be installed prior to fuel load. The bell will be of acceptable audible levels to allow for sufficient alarm response for the operators and also to allow for verbal communication between operators.

The second high priority HED for which deferral was requested (HED SI4-04) involves the testability of indicating status lights on the remote shutdown panel. Justification for not correcting this HED until after fuel load is based upon the adequacy of the start-up test program and the rigid security maintained over this panel. All five lights of concern will be electrically and functionally tested during the Limerick start-up test program. During these tests, it will be verified that the electrical circuits function as designed and that the light bulbs are good. At the completion of the tests, the remote shutdown panel will be de-energized by transferring control back to the main control room panels. At this point, the circuits and bulbs will have been verified as functional. This remote shutdown panel is maintained in a locked room with access controlled by operations shift supervision. Additionally, the access doors to the remote shutdown panel room are monitored by the plant security system. The actual transfer of control and control power to the remote shutdown panel from the control room is annunciated in the control room when any one of the transfer switches is in the emergency position. The remote shutdown panel indicating status lights will have a very high probability of working properly if an emergency occurred that required evacuation of the control room. Based on this information, the staff concludes that the test of the indicating status lights in conjunction with the security measures should provide a low probability of burned out bulbs.

The staff finds the correction of high priority HEDs prior to fuel load acceptable. The reasons given for the deferred correction of HED SI4-04 and the planned actions to ensure a high availability of the indicating status lights on the remote shutdown panel are acceptable to the staff.

In our evaluation of the HEDs defined in the applicant's Final Report, we noted that four HEDs were held out for further review. Of these HEDs, two were not rated for safety significance because their earlier assessment had shown that they were of low priority. For the remaining two HEDs, the staff conducted a phone conference with the applicant on August 16, 1984, and requested the priority rating for HED SD3-15 and I5-04. PECO informed the staff that HED SD3-15 had a priority of 4 (no significant effect on operation) and HED I5-04 had a priority of 2 (HEDs that have caused problems or appear likely to cause problems during normal and off-normal operations that could not result in unsafe operations). PECO also stated that these results would be presented in a supplement to the Final Report. Based on the value and significance of these priority ratings, the staff finds this acceptable.

18.1.8 Selection of Design Improvements

Supplement 1 to NUREG-0737 requires the selection of control room design improvements that will correct significant HEDs. It also states that improvements with an enhancement program should be done promptly.

LLNL's TER concludes that the Limerick Summary Report has failed to provide an ample description of any "method" employed in the selection of improvements. A second on-site audit of the DCRDR was conducted on August 8-9, 1984 to obtain the information needed to complete the evaluation of methods employed in the selection of improvements. Based on the information collected and reviewed, the staff concluded that adequate compliance with NUREG-0737, Supplement 1 is being achieved.

18.1.9 Verification That Improvements Will Provide Necessary Corrections and Will Not Introduce New HEDs

NUREG-0737 Supplement 1 requires licensees/applicants to verify that each selected design improvement will produce the necessary correction, and can be introduced in the control room without creating any unacceptable human engineering discrepancies because of significant contribution to increased risk, unreviewed safety questions, or situations in which a temporary reduction in safety could occur.

LLNL's technical evaluation of the applicant's Summary Report concludes:

- The specific process used to verify that selected design improvements will provide necessary correction was inadequately explained in the Limerick Summary Report. In this regard, the report lacks specificity and detail.

- That portion of the Limerick Summary Report which was supposed to address the process whereby new design improvements would be verified not to introduce new HEDs was inadequate. To say that this will be accomplished via "walk-throughs" is sufficiently vague to require more information. We cannot determine whether the requirements of NUREG-0737 are being met.

A second on-site audit of the DCRDR was conducted on August 8-9, 1984 to obtain the information needed to review this issue. Based on the information collected during the audit and the review of the supplementary information, the staff concluded that the intent of NUREG-0737, Supplement 1 is being met.

18.1.10 Coordination of Control Room Improvements With Other Programs

Supplement 1 to NUREG-0737 requires that control room improvements be coordinated with changes from other programs; e.g., safety parameter display system (SPDS), operator training, Regulatory Guide 1.97, and emergency operating procedures (EOPs). LLNL reviewed the applicant's Final Report for coordination of related activities and concluded that it was inadequate to meet the intent of Supplement 1 to NUREG-0737. The staff evaluated the coordination activities during a second on-site audit of the DCRDR on August 8-9, 1984. Based on this evaluation, the staff felt that PECO should provide a detailed description of how the coordination process and method is executed. However, during the earlier In-Progress Audit, the staff did witness coordination among the DCRDR, the Emergency Procedure Guidelines and the SPDS. Based on this evidence, it is appropriate for the applicant to continue with the DCRDR and to provide a detailed description in the next supplement to the Summary Report.

18.1.11 Staff's Conclusions on DCRDR

The staff concludes that the Philadelphia Electric Company's Limerick Unit 1 DCRDR is meeting the NUREG-0737, Supplement 1 requirements for work completed, but the DCRDR is incomplete. The incomplete portion of the DCRDR consists of:

- The use of function and task analysis to identify control room operator tasks and information and control requirements during emergency operation,

- A comparison of display and control requirements with a control room inventory to identify missing displays, and

- Elements of the control room survey to identify deviations from accepted human factors principles.

As the DCRDR is incomplete, the staff has included conditions in the license to address its completion, as stated earlier in this section.

Further, because of insufficient information, the staff was unable to determine if the intent of the coordination of control room improvements with other programs is being met. We request that a detailed description of coordination activities be provided by the applicant in the next supplement to the Summary Report.

The staff's review of the applicant's results in completing the DCRDR will be reported in a further report.

18.2 Safety Parameter Display System

All licensees and applicants for an operating license are required to provide a Safety Parameter Display System (SPDS) that is located convenient to the control room operators in response to Generic Letter 82-33, December 3, 1982 and to the requirements stated in NUREG-0737, Supplement 1. The purpose of the SPDS is to provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. The applicant shall prepare a written safety analysis describing the basis for the selected parameters. Such analyses, along with the specific implementation plan for the SPDS will be reviewed by the staff. The implementation plan should contain schedules for design, development, installation, and full operation of the SPDS as well as a verification and validation plan.

18.2.1 Background

The purpose of the SPDS is to provide a concise display of critical plant variables to control room operators to aid them in rapidly and reliably determining the safety status of the plant. NUREG-0737, Supplement 1 requires licensees and applicants to prepare a written safety analysis describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which include symptoms of severe accidents. Licensees and applicants shall also prepare an Implementation Plan for the SPDS which contains schedules for design, development, installation, and full operation of the SPDS as well as a design Verification and Validation Plan. The Safety Analysis and the Implementation Plan are to be submitted to the NRC for staff review. The results from the staff's review are to be published in a Safety Evaluation Report (SER).

The applicant's response (Reference 10) to Generic Letter 82-33 states that the Limerick SPDS design is included in the Emergency Response Facility Data System (ERFDS). This system is based on the General Electric Emergency Response Information System. We understand that NEDE-30284-P, "Licensing Topical Report for the General Electric Emergency Response Information System," November 1983, defines the SPDS's design for the Limerick SPDS.

The SPDS at Limerick will be part of the Emergency Response Facility Data System. The staff audited the design of General Electric's Emergency Response Information System during July 24-26, 1984. The results from this activity will be published in an audit report. Also, an SER on this generic SPDS is being prepared and will be published in the near future. Where applicable, these results and evaluations from the audit were used by the staff in assessing Limerick's SPDS.

In response to the requirements (NUREG-0737, Supplement 1), for an SPDS, the applicant submitted a Safety Analysis on parameter selection (Ref. 11) for review by the staff. The staff evaluated the applicant's Safety Analysis (Ref. 11) and concluded insufficient information existed to conduct a review. The staff's request for additional information is documented in Reference 12. The applicant's response to this request is contained in Reference 13. The results from the staff's evaluation to date of the Limerick SPDS are presented in the text which follows.

18.2.2 SPDS Description

The Limerick SPDS is a subsystem within the Emergency Response Facility Data System (ERFDS). This system is based on the General Electric Emergency Response Information System.

General Electric has developed an Emergency Response Information System (ERIS), a display system which contains the SPDS function. ERIS is a computer based system and consists of three subsystems which are a Data Acquisition System (DAS), a Data Processing System (DPS), and Data Output Peripherals (DOP). DAS gathers signals and converts these signals into a form usable by a digital computer. The DPS prepares the signals for display upon CRTs and also stores the processed signals for later use. The DOP contain CRTs for the display of plant data. Keyboards are also provided as an operator interface to the display system.

General Electric states that ERIS is based upon the symptom oriented Emergency Procedure Guidelines (EPGs). In the control room, ERIS assists the operating personnel in their functions by displaying the following information on CRTs:

- Real-time plant status to aid in early emergency procedure entry condition recognition. This can be displayed continuously and is monitored by control room operators during normal operations,
- Data to assist the operator in following the emergency procedures including current readings, trends of control parameters, and status of major systems,
- Two-dimensional limits as defined in the emergency procedures. This assists the operator by precluding the need to perform manual calculations to determine margins to limits and graphically showing trends of parameters,
- Critical parameter validation status,
- Critical variable trend plots.

18.2.3 Parameter Selection

Section 4.1f of Supplement 1 to NUREG-0737 states that:

"The minimum information to be provided shall be sufficient to provide information to plant operators about:

(i) Reactivity Control

(ii) Reactor core cooling head removal from the primary system

(iii) Reactor coolant system integrity

(iv) Radioactivity control

(v) Containment conditions."

For review purposes, these five items have been designated as Critical Safety Functions.

The selection of the SPDS display parameters was made by GE based on the BWR generic Emergency Procedure Guidelines (EPGs) (Reference 14). We have confirmed that the parameters selected are consistent with the presently approved BWR EPGs (Revision 3) with one exception. Revision 3 contains a Radioactivity Release Control Guideline which contains an Entry Condition based on off-site radioactivity release rate. The GE basic SPDS display does not contain a monitored parameter dealing directly with radiation measurement.

The SPDS parameters and their relationship to the Critical Safety Functions are summarized in the attached Table 1. The grouping was made by the staff based on inspection of the first level SPDS display format and information furnished by GE at the Design Verification Audit for the GE SPDS. GE has grouped the individual parameter identification to coordinate with the generic EPGs which include separate sequential procedural steps identified under the general concepts of Reactor Pressure Vessel Control and Containment Control. These individual groupings are used for second-level display formats on the GE SPDS. The applicant has chosen to supplement the GE SPDS with a Radiation Meteorological Monitoring System (RMMS) which will display process, stack and area effluent radiation data to serve as a monitor for the Radioactivity Control Critical Safety Function. We find this acceptable, but recommend the monitoring of Containment Radiation for conditions when the effluent paths are isolated.

Neutron flux is a fundamental parameter for monitoring the status of the Reactivity Control Critical Safety Function. An indication of reactivity control should be provided for all power ranges. The GE SPDS provides monitoring of the power level of Average Power Range Monitors (APRMs) during power Operation. For conditions below the APRM range, the GE SPDS does not monitor power level, but does provide scram status.

GE has stated that the combination of power level and scram status is sufficient for monitoring the Reactivity Control Critical Safety Function. It is our understanding based on discussions with GE that following a reactor scram and a core-wide verification of rods in status, the scram status indicator on the SPDS will display "rods in." This display message will not change unless a rod is withdrawn or drifting, in which case the display changes to an alarm (red) indication. Also, in the startup mode, the Intermediate Range Monitor

(IRM) upscale trip results in a rod withdraw block which will result in a scram displayed on the SPDS if a high-high setpoint (120/125 of scale) is exceeded. During some plant conditions, such as performance of core alterations (e.g., fuel loading), if a signal from the neutron monitoring system exceeds a Source Range Monitor (SRM) high-high setpoint, this condition would be indicated on the SPDS scram status indicator. The staff concludes that since the scram signals are directed to the SPDS display, the combination of the APRMs and scram status indicator provides adequate monitoring of the Reactivity Control Critical Safety Function. The staff also recognizes that during periods of startup and heatup, a portion of the plant operations staff would have attention focused on the neutron instrumentation in the control room.

This acceptance is conditional, subject to confirmatory documentation by GE of the preceding information regarding the scram status indicator on the SPDS. In particular, GE should:

1. Document the neutron monitor signals which are directed to the scram status indicator to produce a "scram initiated" message. Discuss the differences in the scram status display during startup, shutdown and refueling conditions.

2. Verify that the information used to generate a "rods in" message is continuously monitored.

This information may be provided as part of the report to be submitted for confirmatory staff review, as identified in the Design Verification Audit Report for the GE SPDS (Reference 15).

We have verified that the GE ERIS design includes sufficient capacity for expandability so that additional parameters (such as hydrogen concentration) may be added as a result of future revisions to the generic EPGs.

The staff finds that the parameter selection for the Limerick SPDS would be acceptable subject to confirmatory staff review of the information identified in this report, and the addition of a Containment Radiation Monitor to identify the status of the Radioactivity Control Safety Function during periods when the containment is isolated.

18.2.4 Display Data Validation

The staff reviewed PECO's response (Ref. 13) to an information request to determine that means are provided in the displays design to assure that the data displayed are valid. Limerick states (Ref. 13) that the validation method used in their SPDS is identical to the method used in all General Electric's SPDSs. The staff recently audited this design and found that the top level display format of critical plant variables contains each plant variable used as entry variables to the Emergency Operation Procedures. This data was presented as numerical data enclosed by a color coded status box. The color code of the status box informs the operator on the validation status of the enclosed data.

The staff's audit determined that as part of the real-time processing of the data, the ERIS/SPDS performs the following checks on analog and digital inputs: comparison of redundant signal, range check, zero adjust, density correction, reference leg boiling check, temperature compensation and an instrument power check in performing data validation. Furthermore, secondary display formats which contained detailed data on the intermediate steps of the data validation process were available for each entry variable to the Emergency Operations Procedures. Properly implemented in a plant, the staff believes this intermediate data should prove valuable to a supervisor in evaluating the validity of the data for use in decision making tasks during emergencies.

Based on the information obtained during our audit of General Electric's SPDS, the staff confirms that means are provided in the Limerick SPDS design to assure that the data displayed are validated.

18.2.5 Human Factors Program

The staff evaluated Philadelphia Electric Company's response (Ref. 13) to an information request for a commitment to a Human Factors Program the development of the SPDS and concluded that the results from our Design Verification Audit of the General Electric SPDS were applicable.

During our Design Verification Audit, we were told that General Electric had hired ANACAPA SCIENCES INC. to conduct a human factors review of selected SPDS display formats. The staff evaluated a report titled "Human Factors and Performance Evaluations of the Emergency Response Information System (ERIS)," July 10, 1984, ANACAPA SCIENCES INC. We found the report to be comprehensive in its scope of review and in the reporting of results, both positive and negative, and in the recommendations made as a result of the evaluation. We evaluated several of the recommendations and noted that many had been implemented into the design.

The staff evaluated the design effort's consistency in the application of colors in the various display formats. This evaluation effort focused upon the RPV CONTROL -- NR/TEMP display format and the CONTAINMENT CONTROL -- NR display format. The initial explanation of how color was used to highlight and code information in these display formats left the staff confused. The staff was concerned that a confusing, complicated application of color would result in operator errors.

To understand the issue, the staff requested a clarification of color coding in terms of the individual data sets for the selected display formats. After considerable explanation by General Electric, it appeared that a logical, consistent application of color had been made. To confirm this judgment, the staff requested that General Electric document how color is used to code information and submit this documentation to the staff for confirmatory review. The Limerick SPDS will also be subject to the results of the staff's confirmatory review.

During the staff audit of General Electric's SPDS design, we evaluated some of the display formats within the system. For the most part, we found the majority of the display formats to be uncluttered and easy to comprehend. However, we found two display formats to be very dense with information. These display formats were the RPV CONTROL -- NR/TEMP display format and the CONTAINMENT CONTROL -- NR display format. Part of the displayed data contained the status of plant systems. General Electric stated that this data was not a part of the SPDS requirements. The staff acknowledged this fact, however, we noted that the operational status of these systems did impact the process variables displayed, which are part of the SPDS and thus represented good integration of related data. We also noted that the individual data sets contained in the display format were not labeled. In times of stress, this could prove difficult for the operator in locating, comprehending and using the information within the display format. The staff recommended that this potential problem be monitored by General Electric during the forthcoming validation tests and during installed operation of the display system.

In evaluating Limericks's response (Ref. 13) to the staff's information request, we noted that the RPV CONTROL -- NR/TEMP display format and the CONTAINMENT CONTROL -- NR display format had been modified and made plant specific to Limerick. These display formats did not contain data on the status of plant systems. The revised display formats are easier to read and comprehend, and the information density is now consistent with that of other display formats in the system.

Based on the information obtained during our audit of General Electric's SPDS and our evaluation of the applicant's responses (Ref. 13), the staff confirms that the applicant did commit to a Human Factors Program in the design of the SPDS. However, we have requested from General Electric additional data for confirmatory staff review on color coding of information in the display formats. The Limerick SPDS will also be subject to the results of the staff's confirmatory review.

18.2.6. Electrical and Electronic Isolation

We reviewed the isolation devices for the Limerick SPDS and conclude they are identical to the isolation devices used in the General Electric generic SPDS which the staff found them to be acceptable isolation devices.

18.2.7 Verification and Validation Plan

In the applicant's response to the information request, the applicant states that the Verification and Validation Program used in the development of the Limerick SPDS is identical to the Program generically used for all GE supplied SPDSs. During the Design Verification Audit of General Electric's SPDS, the staff evaluated the Verification and Validation Program (V&V) used in the design of the system. General Electric described the V&V Program and stated that it was patterned after NSAC-39*. The staff audited specific design verification activities and requested General Electric to demonstrate how a problem defined from verification activities was resolved. The staff also evaluated the ERIS Test Requirement Document which is being used by General Electric to prepare for the Validation Test of the system.

The staff found the General Electric Verification and Validation Program to be similar to the one described in NSAC-39. In evaluating the application of the V&V Program, we found that General Electric was able to demonstrate how staff selected problems, were documented and adequately resolved. In evaluating the ERIS Validation and Test Requirement Document, the staff did successfully

*NSAC-39, "Verification and Validation for Safety Parameter Display Systems," December 1981, Nuclear Safety Analysis Center, Electric Power Research Institute.

correlate test requirements with the functional requirements of the design. We also learned that Validation Test Procedures are currently being prepared by General Electric. The Validation Test for ERIS are to be conducted late this year (1984) with a test report on results due by February, 1985. Based on the staff's review of the Verification and Validation Program, and of its application in the design process, the staff concludes the program is adequate for the design of the SPDS and is being effectively used in the development of the system.

18.2.8 Unreviewed Safety Questions

In Reference 13, the applicant defines conclusions regarding unreviewed safety questions or changes to technical specifications. The applicant states that the implementation of the Limerick SPDS does not involve an unreviewed safety question or require a change in the Limerick Technical Specifications. Based on the Commission approved requirements in NUREG-0737, Supplement 1, the staff concludes the applicant may continue to implement the SPDS.

18.2.9 Implementation Plan

In Reference 9, the applicant defines the current status of the SPDS and discusses an implementation schedule. The applicant states all system hardware is presently installed and powered up. The display system is undergoing calibration and debugging. The display formats for the SPDS are functional and can be called up in the Control Room, TSC, and EOF. Operator training on use of the system is complete. Additional time will be required to complete the debugging process and until completed, the ERFDS and SPDS cannot be considered functional. PECO proposes the following schedule for operation of the ERFDS and SPDS:

- Hardware Installed and Powered	Complete
- SPDS Display Formats Loaded into ERFDS	Complete
- Operator Training	Complete
- SPDS Displays Functional	March 1, 1985
- Reg. Guide 1.97 Displays Functional	April 1. 1985

During the Design Verification Audit of General Electric's generic SPDS, the staff learned that Validation Tests for the system are to be conducted late in 1984, with a test report on results by February 1985. As the Limerick SPDS is a General Electric generic SPDS, the staff believes it is prudent to wait until the results of the Validation Tests have been assessed prior to the operational use of the SPDS displays. Thus, an April 1, 1985 operations date for these systems is acceptable to the staff. The staff will condition the license accordingly.

18.2.10 Staff's Conclusions on SPDS

The NRC staff reviewed the Philadelphia Electric Company's Limerick Safety Analysis and response to an information request to confirm the adequacy of the parameters selected to be displayed to monitor critical safety functions, to confirm that means are provided to assure that the data displayed are valid, to confirm that the applicant has committed to a Human Factors Program to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator and to confirm that the SPDS is suitably isolated.

Based on its review, the staff confirms that:

- means are provided in the SPDS design to assure that the data displayed are valid,
- an appropriate commitment to a Human Factors Program was made in the design of the SPDS.
- the SPDS will be suitably isolated from electrical and electronic interference with equipment and sensors that are used in safety systems.

The staff finds the parameter selection of the Limerick SPDS would be acceptable subject to:

- the confirmatory staff review of information requested from General Electric on the scram status indicator within the SPDS,
- The addition of a Containment Radiation Monitor to identify status of the Radioactivity Control safety Function during periods when containment is isolated.

The implementation of the SPDS at Limerick is incomplete. Accordingly, the staff will condition the license to require the Limerick SPDS to be operational and functionally available for use by operators no later than March 30, 1985.

Table 1 SPDS Safety Parameters Limerick Generating Station

Critical Safety Function	Parameter
Reactivity Control	APRMS Scram Status (All rods in)
Reactor Core Cooling and Heat Removal	Reactor Vessel Water Level Reactor Vessel Pressure Reactor Vessel Water Temperature Trend Plot
Reactor Coolant System Integrity	Reactor Vessel Pressure Reactor Vessel Isolation Status Drywell/Containment Pressure
Containment Integrity	Containment/Drywell Temperature Drywell Pressure Suppression Pool Water Level Suppression Pool Water Temperature Suppression Pool Makeup System Status Containment Isolation Status
Radioactivity Control*	Plant Stack Radiation Monitors Area Radiation Monitors Process Effluent Radiation Monitors

*The identified parameters are part of an RMMS display separated from the generic SPDS display

Section 18 References

1. Letter to A. Schwencer, NRC, from J.S. Kemper, Vice-President, Philadelphia Electric Company, subject: Limerick Generating Station, Units 1 and 2 Control Room Design Review, August 31, 1983, with enclosure titled: Detailed Control Room Design Review Program for Philadelphia Electric Company's Limerick and Peach Bottom Plants.

2. Memorandum for Thomas M. Novak, NRC, from William T. Russell, NRC, subject: Response To Limerick Program Plan Submittal, November 16, 1983.

3. Memorandum for Albert Schwencer, NRC, from V.A. Moore, NRC, subject: Results From NRC's In-Progress Audit of Limerick's Detailed Control Room Design Review (DCRDR), June 12, 1984.

4. Memorandum for Voss A. Moore, NRC, from S.H. Weiss, NRC, subject: Meeting Summary--Task Analysis Requirements of Supplement 1 to NUREG-0737, May 4, 1984 Meeting With BWR Owners' Group Emergency Procedure Guidelines and Control Room Design Review Committees, May 4, 1984.

5. Letter to A. Schwencer, NRC, from J.S. Kemper, Philadelphia Electric Co., subject: Limerick Control Room Design Review Final Report, June 25, 1984, with attachment: Philadelphia Electric Company, Limerick Plant, Control Room Design Review, Final Report, June 1984.

6. Memorandum for Thomas M. Novak, NRC, from William T. Russell, NRC, subject: Request for Meeting with Limerick to Resolve a Concern on the Detailed Control Room Design Review (DCRDR), July 11, 1984.

7. Minutes of August 7 meeting with PECO.

8. Letter to A. Schwencer, NRC, from J.S. Kemper, Philadelphia Electric Company, subject: Limerick Generating Station Units 1 and 2, Limerick Control Room Design Review, August 16, 1984.

9. Letter to A. Schwencer, NRC from J.S. Kemper, Philadelphia Electric Company, subject: Limerick Generating Station, Units 1 and 2, Deferral of Certain Pre-operational Tests and Construction Completion Items Unit 1 After Fuel Load, July 17, 1984.

10. Letter to Darrell G Eisenhut, NRC, from V.S. Boyer, Philadelphia Electric Company, subject: Limerick Generating Stations, Units 1 and 2, Reference: Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability, Generic Letter No. 82-33, April 15, 1983.

11. Letter to A. Schwencer, NRC, from John S. Kemper, Philadelphia Electric Company, subject: Limerick Generating Station, Units 1 and 2, Safety Parameter Display System (SPDS), September 2, 1983, with attachment: Safety Analysis for Parameter Selection for Safety Parameter Display System (SPDS).

12. Memorandum for A. Schwencer, NRC, from V.A. Moore, NRC, subject: Request for Additional Information Concerning the Limerick SPDS, Review Data, April 24, 1984.

13. Letter to A. Schwencer, NRC, from J.S. Kemper, Philadelphia Electric Company, subject: Limerick Generating Station, Units 1 and 2, Request for Additional Information Limerick SPDS Review, July 20, 1984.

14. Letter, T. J. Dente (BWR Owners Group) to D. G. Eisenhut (NRC) dated December 22, 1982, transmitting BWR Emergency Procedure Guidelines, Revision 3 (dated December 8, 1982).

15. Design Verification Audit Report for the General Electric Safety Parameter Display System (to be published).

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards issued an Interim Report on Limerick dated October 18, 1983 in which it indicated it wished to return to the subjects of emergency planning, plant security, severe seismic events, consequences of cooling tower failure and probabilistic risk assessment.

The current evaluation of the emergency planning reviews is addressed in Section 13.3 of Supplement 3 to the SER. The current evaluation of the physical security review is addressed in section 13.6 of Supplement 2 to the SER. The evaluation of seismic events more severe than the safe shutdown earthquake and the evaluation of the applicant's probabilistic risk assessments are addressed in the staff's report "Review Insights on the Probabilistic Risk Assessment for the Limerick Generating Station," NUREG-1068, and in the Final Environmental Statement, NUREG-0974.

A summary of the staff's evaluation of the effects of cooling tower failures follows. In response to this issue the applicant submitted additional information in a letter dated January 18, 1984 to the NRC staff. This issue was also addressed in response to concerns raised by the Atomic Safety and Licensing Board on the margins of structural capability of Category I structures to resist blast overpressure and on the mode of structural failure of the cooling towers. The applicant prepared and presented to the Board testimony pertinent to these concerns by letter of M. J. Wetterhahn dated February 28, 1984. The applicant's testimony is essentially consistent with their letter to the staff dated January 18, 1984. The staff has reviewed the applicant's testimony and in addition to testimony filed by the staff by letter of February 17, 1984, the following is a summary of the staff's evaluation of the applicant's testimony.

As noted in the staff's testimony the staff reviewed and accepted the basis of the blast overpressure considered appropriate for Limerick which is assumed to result from detonation of an explosive cargo in a railroad boxcar.

For the analysis of structural resistance of safety-related structures to the blast overpressure, the applicant has used a methodology contained in the Department of the Army Technical Manual "Structures to Resist the Effects of Accidental Explosions," (TM 5C--1300). This methodology is based on plastic behavior of the material and assumes that the ductility of a structure can be mobilized to absorb the energy resulting from the explosion. Ductility of a structure is defined in terms of the ratio of plastic deformation to the deformation at its elastic limit. The applicant has used a ductility ratio of 3 for reinforced concrete structures, which is acceptable to the staff.

The applicant has also evaluated the global effects on safety-related structures, specifically overturning and story shear of the building due to blast overpressure. It was found that the moments and shears are smaller than those caused by the safe shutdown earthquake (SSE) for which these structures are designed. It is, therefore, concluded by the applicant that these global effects are not the controlling parameters. The staff has reviewed the applicant's procedure and found it to be appropriate. The staff concurs with the applicant's conclusion.

The applicant also examined the modes of failure of the cooling tower. There are two possible modes of failure: one is overturning and the other, collapse of the cooling tower. The category I structure nearest to the cooling tower is the spray pond pumphouse which is 520.5 ft. from the base of the cooling tower, a distance greater than the height (507.5 ft) of the cooling tower. Thus, assuming that the cooling tower rigidly rotates about its base, which is very unlikely, the cooling tower will not impact the pumphouse. For the case of cooling tower collapse, the applicant assumed that a concrete fragment, of dimensions 5 ft x 5 ft x 1 ft, falls freely from the highest point of the cooling tower. The applicant calculated a penetration of 2.8 ft into the ground. This penetration is less than the 4 ft soil cover or equivalent protection for the safety-related buried piping and electric duct banks; therefore, the applicant concluded that these safety-related buried items will not be damaged. The staff has reviewed the applicant's assessment, found it reasonably conservative and concurs with the applicant's conclusion.

On the basis of staff's review and evaluation of the applicant's testimony, the staff has concluded that the two structural engineering issues resulting from blast overpressure and cooling tower failure have been satisfactory resolved. The ASLB's findings on this issue may be found in its Partial Initial Decision issued on August 29, 1984 at pages 56-76.

APPENDIX A

CHRONOLOGY
LIMERICK GENERATING STATION, UNITS 1 AND 2

September 11, 1984	Letter from applicant on extreme wind hazard
September 12, 1984	Letter from applicant on III.D.1.1, Primary Coolant Outside Containment
September 12, 1984	Letter to applicant on IDVP
September 12, 1984	Letter from applicant on containment negative pressure design limit
September 13, 1984	Letter from applicant on Technical Specification surveillance requirement 4.6.1.4
September 14, 1984	Letter from applicant on offsite dose calculation manual
September 14, 1984	Letter from applicant on primary containment isolation valves
September 20, 1984	Letter to applicant on test abstracts
September 20, 1984	Letter from applicant on term of license
September 20, 1984	Letter from applicant on the ISEG
September 21, 1984	Letter to applicant on technical specifications
September 21, 1984	Letter from applicant on question 440.5
September 21, 1984	Letter from applicant on exemptions from GDC 56 for isolation valves
September 21, 1984	Letter from applicant on exemption from GDC 61 for SGTS
September 21, 1984	Letter from applicant on drywell/suppression chamber vacuum breaker valve position switches
September 21, 1984	Letter from Torrey Pines Technology on IDVP
September 24, 1984	Letter from applicant on extreme wind hazard
September 25, 1984	Letter from applicant on IDVP
September 26, 1984	Letter from applicant on liquid nitrogen inerting systems

September 26, 1984	Letter from applicant on primary containment isolation valve closure times
September 27, 1984	Letter from applicant on startup test program
September 27, 1984	Letter from applicant on RCS pressure isolation valve leakage testing
September 28, 1984	Letter from applicant in response to staff's September 20, 1984 letter on test abstracts
September 28, 1984	Letter to applicant emergency response plan for Skippack Township
October 1, 1984	Letter from applicant on fuel loading schedule
October 2, 1984	Letter to applicant on draft license
October 2, 1984	Letter from applicant on IDVP review experience
October 3, 1984	Letter to applicant approving offsite dose calculation manual
October 4, 1984	Letter from applicant on thermal-hydraulic stability
October 4, 1984	Letter from applicant on test and construction deferrals
October 4, 1984	Letter from applicant on ISI for feedwater check valves
October 5, 1984	Letter from applicant on startup test procedure
October 9, 1984	Letter from applicant on regulations
October 10, 1984	Letter from applicant on technical specifications
October 10, 1984	Letter from applicant on administrative procedures
October 12, 1984	Letter from applicant on turbine control valves
October 12, 1984	Letter from applicant on turbine system ISI
October 12, 1984	Letter from applicant on III.D.1.1 test results
October 15, 1984	Letter from applicant on scram system piping
October 16, 1984	Letter from applicant on control room design review TER
October 17, 1984	Letter from applicant commenting on draft license
October 19, 1984	Letter from applicant on the ultimate heat sink
October 19, 1984	Letter from applicant on GDC 2 and 4
October 19, 1984	Letter from applicant on ODCM
October 25, 1984	Letter from applicant on GDC-19

APPENDIX H

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This supplement to the Safety Evaluation Report is a product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report. A list of consultants follows the list of staff members.

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APPENDIX N

SAFETY EVALUATION SUPPLEMENT ON PRESERVICE INSPECTION RELIEF REQUEST

I. INTRODUCTION

This section was prepared with technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

For nuclear power facilities whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, 10 CFR 50.55a(g)(2) specifies that components (including supports) which are classified as ASME Code Class 1 and 2 must meet the preservice examination requirements set forth in editions of Section XI of the ASME Code and Addenda in effect six months prior to the date of issue of the construction permit. The provisions of 10 CFR 50.55a(g)(2) also state that components (including supports) may meet the requirements set forth in subsequent Editions and Addenda of this Code which are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein.

In submittals dated July 17, 1984, August 7, 1984, August 23, 1984, August 27, 1984, and August 30, 1984, the Applicant requested relief from ASME Section XI Code requirements which have been determined to be not practical. These relief requests were supported by information pursuant to 10 CFR 50.55a(a)(2)(i). Therefore, the staff evaluation consisted of reviewing these submittals to the requirements of the above referenced Code and determining if relief from the Code requirements were justified.

II. TECHNICAL REVIEW CONSIDERATIONS

A. The Limerick Unit 1 construction permit was issued on June 19, 1974. In accordance with the provisions of 10 CFR 50.55a(g), which allows for the use of subsequent editions and addenda of the Code, the PSI Program (with the exception of the reactor pressure vessel (RPV)) meets the requirements of the 1974 ASME Code, Section XI through the Summer 1975 Addenda as modified by Appendix III of the Winter 1975 Addenda and paragraph IWA-2232 of the Summer 1976 Addenda. The RPV examinations are in accordance with the 1980 ASME Code including the Winter 1980 Addenda.

B. Verification of the as-built structural integrity of the primary pressure boundary is not dependent on the Section XI preservice examination. The applicable construction codes to which the primary pressure boundary was fabricated contain examination and testing requirement which by themselves provide the necessary assurance that the pressure boundary components are capable of performing safely under all operating conditions reviewed in the FSAR and described in the plant design specification. As a part of these examinations, all of the primary pressure boundary full penetration welds were volumetrically examined (radiographed) and the system will be subjected to hydrostatic pressure tests.

C. The benefits of the preservice examination include providing redundant or alternative volumetric examination of the primary pressure boundary using a test method different from that employed during the component fabrication. Successful performance of preservice examination also demonstrates that the welds so examined are capable of subsequent inservice examination using a similar test method. In the case of Limerick Generating Station Unit 1, a large portion of the preservice examinations required by the ASME Code was performed. Failure to perform a 100% preservice examination of welds identified below will not significantly affect the assurance of the initial structural integrity.

D. In some instances where the required preservice examinations were not performed to the full extent specified by the applicable ASME Code, the staff may require that these examinations or supplemental examinations be conducted as a part of the Inservice Inspection (ISI) Program. The ISI Program is based on the examination of a representative sample of welds to detect generic degradation. In the event that the welds identified in the PSI relief requests are required to be examined again, the possibility of augmented inservice inspection will be evaluated during review of the Applicant's initial 10-year ISI Program. An augmented program may include increasing the extent and/or frequency of inspection of accessible welds.

III. EVALUATION OF RELIEF REQUESTS

The Applicant requested relief from specific PSI requirements in a submittal dated July 17, 1984. In submittals dated August 7, 1984, August 23, 1984, August 27, 1984, and August 30, 1984, the Applicant requested relief on other subjects and revised or deleted other requests. These submittals contain descriptions, a detailed list of components for which relief is requested in the Component Summary Table, Revision 1 (Attachment 7 of the August 23, 1984 submittal), a Safety Impact Summary for systems for which relief is requested (Attachment 5 of the July 17, 1984 submittal), and justification of relief requests. Based on the information submitted by the Applicant and review of the design, geometry, and materials of construction of the components, certain preservice requirements of the ASME Boiler and Pressure Vessel Code, Section XI have been determined to be impractical. Imposing these requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(2), conclusions that these preservice requirements are impractical are justified as followed.

Unless otherwise stated, references to the Code refer to the ASME Code, Section XI, 1974 Edition including Addenda through Summer 1975 plus Appendix III of the Winter 1975 Addenda and Paragraph IWA-2232 of the Summer 1976 Addenda.

Relief requests numbers 19 and 20, covering Class 1 Pressure Retaining Welds in Piping (fabrication flaws) and Class 2 Pressure Retaining Welds in Piping (fabrication flaws), were not evaluated because the Applicant has performed additional examination and the Applicant formerly withdraw these requests in a letter dated August 30, 1984.

A. <u>Category B-A, Pressure Retaining Welds in Reactor Vessel and Category B-D, Reactor Vessel Nozzle Weld (Relief Requests Numbers 1 through 5)</u>

Code Requirements:

Examination Category B-A - Table IWB-2500-1 in the Winter 1980 Addenda of Section XI requires a 100% volumetric examination of the subject pressure retaining welds in the reactor vessel.

Examination Category B-D - Table IWB-2500-1 in the Winter 1980 Addenda of Section XI requires a 100% volumetric examination of the subject nozzle weld in the reactor vessel.

Code Relief Request: Relief is requested from performing 100% of the Code-required volumetric examination.

Reason for Request: The design of the reactor vessel includes nozzles that prevent automated ultrasonic examination of limited areas on some of the vessel welds. In most cases these limitations were reduced by applying supplemental manual examinations. The total coverage for almost all of these vessel welds exceeds 90% except for circumferential seam weld AE (86%), longitudinal seam weld BC (81.8%), and bottom head welds DA-DF (84.3%). The bottom head longitudinal (dollar plate) weld seam DG was examined over a length of 18.5 in. at each end. The remainder of this weld is unexaminable due to control rod drive housings. One nozzle (feedwater inlet nozzle N4D) had 83.3% coverage. This reduced coverage was caused by interference from another nozzle.

Staff Evaluation: In a letter dated July 17, 1984, the Applicant provided a detailed analysis of the examination coverage. The staff has reviewed this document and determined that the Applicant has examined the welds to the maximum extent possible. The staff concludes that the large extent of Section XI ultrasonic volumetric examination, the volumetric and surface examinations performed during fabrication, and the hydrostatic test demonstrate an acceptable level of preservice structural integrity.

B. Category B-J, Circumferential and Longitudinal Pipe Welds and Category C-G, Branch Connection Welds (Relief Request Numbers 6, 7, 13, 14, 22)

Code Requriement:

Examination Category B-J - Table IWB-2600 in the Summer 1975 Addenda of Section XI specified a volumetric examination for circumferential and longitudinal pipe welds, and branch pipe connection welds exceeding six inches in diameter.

Examination Categories C-F and C-G - Table IWC-2600 in the 1974 Section XI specifies a volumetric examination for piping circumferencial butt welds, longitudinal weld joints in fittings, and branch pipe-to-pipe weld joints.

Code Relief Request: Relief is requested from performing 100% of the Code-required volumetric examination.

Reason for Requests: The design of Class 1 and Class 2 piping system has welded joints, such as, pipe-to-fitting and pipe-to-component, which physically obstruct all or part of the required Section XI examinations from the fitting or component side of the weld specified. The Applicant has identified the piping system welds with obstructions, identified the obstruction, and estimated the percent of volume coverage for a total of 118 welds with limited coverage in Revision 1 of the Component Summary Table (48 pages) contained in the August 23, 1984 submittal. The July 17, 1984 submittal contains a Safety Impact Summary for the systems which contain the welds with limited coverage.

Staff Evaluation: The staff has determined that the volumetric examination of the subject welds to the extent required by the Code is impractical because of the design of the piping systems. The Applicant has augmented the Section XI preservice ultrasonic examinations by complete liquid penetrant (PT) examinations (except for Class 2 welds RH 190 and RH 194). The PT examinations were performed in accordance with the 1977 Edition of Section XI as modified by the Addenda through Summer 1978. The staff therefore concludes that limited Section XI ultrasonic examinations, the augmented surface examinations, the volumetric examinations performed during fabrication, the hydrostatic test, and the Applicant's Safety Impact Summary demonstrate an acceptable level of preservice structural integrity.

C. Class 1, Category B-K-1 Support Members for Pumps (Relief Request Number 8)

Code Requirement: Table IWB-2600 in the Summer 1975 Addenda of Section XI specifies a volumetric examination for category B-K-1 integrally-welded support attachment welds.

Code Relief Request: Relief is requested from performing the Code-required volumetric examination on six (6) support welds.

Reason for Request: The pump pressure boundary castings material, specification ASME SA-351 GR.CF8M, and wall thickness prevents meaningful results from Section XI ultrasonic examination.

Staff Evaluation: The staff has determined that the fabrication examinations (radiography and surface examinations of the pump pressure boundary castings plus visual examination of all fittings) and the Applicant's Safety Impact Summary demonstrate an acceptable level of preservice structural integrity.

D. Class 1, Category B-L-2 Pump Casings and Category B-M-2 Valve Bodies (Relief Requests Numbers 9 and 10)

Code Requirement:

Examination Category B-L-2 - Table IWB-2600 in the Summer 1975 Addenda of Section XI requires a visual examination of pump casing internal pressure boundary surfaces.

Examination Categories B-M-2 - Table IWB-2600 in the Summer 1975 Addenda of Section XI requires a visual examination of valve body internal pressure boundary surfaces on valves exceeding 4-inch nominal pipe size.

Code Relief Request: Relief is requested from performing the Code-required visual examination of the internal surface on 2 pumps (comprising 1 group) and 69 valves (comprising 17 groups).

Reason for Request: The Applicant states that the integrity of the pump and valve pressure boundaries has been verified by the construction code examination and testing requirements. This included radiography, surface examinations, and hydrostatic pressure tests.

Staff Evaluation: The staff has determined that disassembly of pumps and valves for the sole purpose of performing preservice visual examination is not practical. The staff has reached the conclusion that the construction code examinations and tests exceed the requirements for visual examination and therefore, are an acceptable alternative to the Section XI preservice visual examinations.

E. Class 2, Category C-B Nozzle Welds and Category C-A Pressure Vessel Welds (Relief Requests Numbers 11 and 21)

Code Requirement:

Examination Category C-A - Table IWC-2600 in the 1974 Section XI Code requires volumetric examination of pressure vessel circumferential butt welds.

Examination Category C-B - Table IWC-2600 in the 1974 Section XI Code requires volumetric examination of nozzle-to-vessel welds.

Code Relief Request: Relief is requested from performing 100% of the Code-required volumetric examination on four (4) nozzle-to-RHR heat exchanger welds and five (5) RHR heat exchanger vessel welds.

Reason for Request: Joint configurations and other components prevented 100% examination coverage. The total coverage for all the subject nozzle welds was 90% or greater except for one nozzle which had 80% coverage. For the subject heat exchanger vessel joints, the coverage ranged from 50% to 90%. Details of the limited coverage are in the Component Summary Table contained in the Applicant's August 23, 1984 submittal.

Staff Evaluation: The staff concludes that the large extent of Section XI ultrasonic volumetric examinations, the volumetric and surface examinations performed during fabrication, the hydrostatic test, and the Applicant's Safety Impact Summary demonstrate an acceptable level of preservice structural integrity.

F. Class 2, Category C-C, Integrally-Welded Support Attachments to Vessels and Category C-E-1, Integrally-Welded Support Attachments to Piping (Relief Requests Numbers 12 and 16)

Code Requirement:

Examination Category C-C - Table IWC-2600 in the 1974 Section XI Code requires surface examinations of pressure vessel integrally-welded supports.

Code Relief Request: For examination category C-C and C-E-1, relief is requested from performing 100% of the Code-required survace examination on 15 RHR heat exchanger support attachment welds. For examination category C-E-1, relief is also requested to use the 1980 Edition of Section XI, as modified by the Addenda through the Winter 1981, to define the examination area.

Reason for Request: For the examination category C-C and C-E-1 attachments, the joint configurations and external obstructions prevent access to portions of the required examination area. For the examination category C-E-1, the later approved Code and Addenda eliminates the requirement to examine large sections of pipe adjacent to the supports.

Staff Evaluation: For examination categories C-C and C-E-1, the limited Section XI surface examinations supplemented by the fabrication examinations and the Applicant's Safety Impact Summary demonstrate an acceptable level of preservice structural integrity. Updating to the requirements of later approved Code and Addenda is permitted by the regulation. Thus for examination category C-E-1, the examination areas, as defined by the later approved Code and Addenda, are acceptable.

G. Class 2, Category C-F, Pressure Retaining Welds in Piping (Relief Request Number 15)

Code Requirement: Circumferential butt welds and longitudinal weld joints in pipe fittings included in Code Category C-F of Table IWC-2520 shall be volumetrically examined per Items C.2.1 and C.2.2 of Table IWC-2600. The examination volume includes the weld plus the base metal for a distance of one (1) wall thickness beyond the edge of the weld.

Relief Request: Relief is requested from the base metal volume requirement of Table IWC-2520. Radiography was utilized as the volumetric examination technique. The examination volume included the weld plus the base metal for a minimum distance of ¼ inch beyond the edge of the weld. There are 40 circumferential and 80 longitudinal welds associated with 20 Main Steam elbows included in this relief request. These welds are identified in the Component Summary Table, Revision 1, submitted August 23, 1984.

Reason for Request: Laminar indications, though not rejectable to the applicable Code Section III requirements, precluded ultrasonic testing from providing a meaningful Section XI volumetric examination. The construction radiographs for the welds in question were reviewed and additional radiography was performed to achieve coverage in excess of the requirements of the 1980 Edition of Section XI, including the Addenda through the Winter 1981, per Figure IWC-2500-7 (Anticipated Code Edition for ISI Program). Radiography will be used for subsequent ISI. The preservice volumetric examinations have been augmented by complete liquid penetrant tests, which were performed in accordance with the 1977 Edition of Section XI, including the Addenda through Summer 1978.

The integrity of the piping pressure boundary has also been verified by construction code examination requirements. Shop welds were radiographed in accordance with that edition of ASME Section III in effect at the time of procurement. Field weld examinations, which include radiography and hydrostatic pressure tests, were performed in accordance with the 1974 Edition of Section III including Addenda through Winter 1974.

The preservice examination of the welds in these 20 elbows is divided into 120 volumes. Fifty-six (56) of the volumes were examined ultrasonically (0°, 45° axial, 45° circumferential). A breakdown of the examinations is as follows: Number Complete - 29, Number Incomplete - 27, and Number with Recordable Indications - 21. The majority of these UT examinations were performed from the pipe side due to the laminar indications detected in the elbow base metal during the 0° straight beam examination.

The Applicant states that for inservice inspection, both the required volumetric and surface examination will be performed. However, radiography will be used in lieu of ultrasonic testing. UT will be used for interrogative purposes when possible when a change is noted during comparison of the baseline radiographs with the subsequent inservice inspection radiographs.

Staff Evaluation: The staff has determined that the subject welds contain laminations on the elbow-side of the weld that limits ultrasonic examination. The staff has reached the conclusion that dye penetrant and radiographic fabrication examination, and the additional Section XI radiography supplemented by the limited preservice ultrasonic examination on 56 weld volumes demonstrate an acceptable level of preservice structural integrity.

Although the Code permits a radiographic volumetric examination for inservice inspection, the Applicant has determined that ultrasonic examination can be performed on the pipe side of the weld. For the sample of welds subject to inservice examination, the Applicant should include in the ISI Program, an ultrasonic examination on the pipe-side of the weld and a radiographic examination on the elbow-side of the weld that is effective for the detection of service-induced defects. In addition, a "best effort" ultrasonic examination should be performed on the elbow side of the weld to monitor the condition of the weld to the extent practical. The staff will review the examination of these welds, including appropriate acceptance criteria, during the review of the ISI Program.

H. Class 2, Category C-F, Pressure Retaining Welds in Pumps (Relief Request Number 17)

Code Requirement: Pump casing weld joints included in Code Category C-F of Table IWC-2520 shall be volumetrically examined per Item C.3.1 of Table IWC-2600. The examination volume shall include 100% of the weld plus the base metal for one wall thickness beyond the edge of the weld.

Relief Request:

Relief is requested from examining 100% of the required volume of the C-F welds for reasons noted in the Component Summary Table. There are 40 welds included in this relief request.

Reason for Request: Twenty-four (24) welds on the RHR and Core Spray Pumps received a limited preservice volumetric examination due to joint configurations (i.e., fitting-to-component) and 16 welds are encased in concrete. The Applicant has identified the welds with limited examinations, identified the obstructions, and estimated the percent of volume coverage in Revision 1 of the Component Summary Table contained in the August 23, 1984 submittal. The Applicant also states that inservice inspection of those pump shell welds encased in concrete will be deferred until such time that the pump is removed for maintenance. Visual examinations from the exterior will be performed during system preservice tests. Shell leakage can be detected at the foundation construction joints.

Staff Evaluation: The staff has determined that the limited Section XI examinations supplemented by the fabrication examinations and the Applicant's Safety Impact Summary submittal July 17, 1984 demonstrate an acceptable level of preservice structural integrity.

I. Class 2, Category C-E-2 Support Components for Pumps (Relief Request Number 18

Code Requirement: Table IWC-2600 in the 1974 Section XI Code requires visual examination for pump support components.

Code Relief Request: Relief is requested from visually examining the portion of pump anchor bolting that is encased in concrete for a total of 88, 1-1/4 inch nominal diameter bolts, comprised of 10 bolts on each of 4 RHR pumps, 10 bolts on each of 4 core spray pumps, and 8 bolts on the HPCI pump.

Reason for Request: Portions of the bolts are not accessible for visual examination.

Staff Evaluation: The fabrication examinations included visual examination of the threads, shanks, and heads (were applicable) plus surface examination on either the finished bolting studs or just prior to threading. Also, the accessible portions of the installed bolting received the Section XI required visual examination and final torque settings were checked. The staff concludes that the above examinations and checks are an acceptable alternative to the Code-required visual examination and therefore demonstrate an acceptable level of preservice structural integrity.

However, for inservice examination, the Applicant should consider use of ultrasonic examination to cover the portion of the anchor bolting not accessible to visual examination as recommended in NUREG-0943, "Threaded Fastener Experience in Nuclear Power Plants," dated January 1983. The staff will evaluate the possibility of volumetric examination of these bolts during review of the inservice inspection program.

J. Class 2, Category C-D, Pressure Retaining Bolting (Exceeding 1-in. Diameter) for Pumps and Valves (Relief Request Numbers 23 and 24)

Code Requirement: Table IWC-2600 in the 1974 Section XI Code requires visual examination and either surface or volumetric examination for the ubject examination of Category C-D bolting.

Code Relief Request: Relief is requested from performing the visual examination of 38, 1-3/4 inch nominal diameter HPCI Pump impeller casing bolts and 100% of the required surface examinations of 9 bolting sets (104 bolts, all less than 2 inch nominal diameter) for valves as listed in the Component Summary Table included in the Applicant's August 23, 1984 submittal.

Reason for Request: Disassembly would be required to complete all the required visual examinations.

Staff Evaluation: For the pump and valve bolting, visual and surface examinations were performed during fabrication. ASME Section XI ultrasonic examination was performed on the pump studs. Section XI surface examination, using the magnetic particle method, was performed on the accessible areas of the valve studs, with coverage from 75% to 92% of the Code required surface. The staff has determined that disassembly of pumps and valves for the sole purpose of performing preservice visual examination is not practical. The staff concludes that the above examinations are an acceptable alternative to the Code-required visual examination and therefore demonstrate an acceptable level of preservice structural integrity.

K. Class 1, Category B-J, and Class 2, Categories C-F and C-G, Pressure Retaining Welds in Piping (Relief Requests 25 and 26)

Code Requirement: Those pipe circumferential pressure retaining welds included in Code Category B-J of Table IWB-2500 shall be volumetrically examined per Item No. B4.5 of Table IWB-2600. Those pipe circumferential pressure retaining welds included in Code Categories C-F and C-G of Table IWC-2520 shall be volumetrically examined per Item No. C2.1 of Table IWC-2600. The following data is required to be recorded to document the examinations per subarticle III-4500:

a. data sheet identify and date;
b. examination personnel;
c. applicable calibration sheet identity;
d. examination procedure and revision;
e. surface from which examination was conducted;
f. record of indication (or lack of) which includes search unit location and orientation applicable to reflector; peak amplitude, reference level, and end points at reference level (parallel to reflector) along with the minimum and maximum sweep readings to the reflector;
g. date and time period of the examination.

Relief Request: Relief is requested from the recording requirement of item III-4500(g) as applied to geometric reflectors. There are 37 Category B-J welds and 70 Categories C-F and C-G welds included in this relief request. These welds are identified in the Component Summary Table, Revision 1, submitted August 23, 1984.

Reason for Relief: For geometric reflectors, the information not recorded on a consistent basis was the circumferential location (L) of the search unit relative to the zero datum for the peak amplitude response. Inside diameter root geometry which was recorded as "intermittent 360°" can be confirmed by data plots and/or review of the ASME Section III radiographs. Fifty-nine of the Categories C-F and C-G welds will not require volumetric examination during inservice inspection. There is no impact on plant safety as a result of this relief request.

Staff Evaluation: The staff has concluded that this relief request is acceptable for PSI because the applicant has determined that the reflectors are geometric in origin and, therefore, recording the specific location of the peak amplitude response for geometric reflectors has no impact on plant safety. However, the Code requirements for recording indications should be followed during inservice examinations.

IV. CONCLUSIONS

Based on the foregoing, pursuant to 10 CFR 50.55(a)(2), certain Section XI required preservice examinations are impractical, and compliance with the requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

The staff technical evaluation has not identified any practical method by which the existing Limerick Generating Station Unit 1 can meet all the specific preservice inspection requirements of Section XI of the ASME Code. Requiring compliance with all the exact Section XI required inspections would delay the startup of the plant in order to redesign a significant number of plant systems, obtain sufficient replacement components, install the new components, and repeat the preservice examination of these components. Examples of components that would require redesign to meet the specific preservice examination provisions are the reactor vessel and a number of the piping and component support systems. Even after the redesign effort, complete compliance with the preservice examination requirements probably could not be achieved. However, the as-built structural integrity of the existing primary pressure boundary has already been established by the construction code fabrication examinations.

Based on the review and evaluation of the cited information, the staff concludes that the public interest is not served by imposing certain provisions of Section XI of the ASME Code that have been determined to be impractical. Pursuant to 10 CFR 50.55a(a)(2), relief is allowed from these requirements which are impractical to implement and would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

APPENDIX O

TECHNICAL EVALUATION REPORT
LIMERICK GENERATING STATION - ULTIMATE HEAT
SINK EXTREME WIND HAZARD ANALYSIS

Dr. E. Simiu

National Bureau of Standards

Technical Evaluation of Report NUS-4507

"Limerick Generating Station - Ultimate Heat Sink Extreme Wind Hazard Analysis"

(March 1984)

Emil Simiu

1. INTRODUCTION

The objective of this evaluation is to assess the validity and the degree of conservatism of the assumptions, data, and mathematical approach used in the Report NUS-4507[a] to estimate the extreme wind hazard to the Ultimate Heat Sink (UHS) of the Limerick Generating Station (LGS).

For the reader's convenience, the following material is excerpted from the Report:

> "The role of the ultimate heat sink (UHS) at Limerick Generating Station (LGS) is to ensure that the temperature of the emergency service water (ESW) and residual heat removal service water (RHRSW) does not exceed the design temperature. Both the ESW and RHRSW systems are required to safely shut down the reactor in the event of a loss of offsite power or an accident. A prolonged loss of the ESW and RHRSW functions could under these conditions lead to core melt and exceedance of 10 CFR 100 limits.

[a] Hereinafter referred to as the Report.

"The ultimate heat sink at Limerick is a spray pond for which it has been demonstrated that under the most conservative design condition there is a 10% margin in thermal performance (PECo, Section 9.2.6*). The spray pond is normally in the standby mode, and is designed to automatically supply water to the ESW and RHRSW systems when required.

"The spray pond has four spray networks, each network having a 50% capacity for shutdown of two units. While all other parts of the ESW and RHRSW systems are protected by barriers from the effects of design basis tornado missiles, the spray pond networks themselves and the feeder pipes feeding those networks are not, and are hence vulnerable to damage.

"Loss of the spray pond networks as a heat sink for the ESW and RHRSW systems does not, however, lead to unavailability of those systems since the pond itself and the cooling towers can be used as heat sinks as a result of operator actions using protected equipment powered from safeguard buses. These realignments of the systems can be initiated from the control room and all the necessary valves and pipework are protected from design basis tornado missiles. The cooling towers themselves, however, are not designed to withstand the wind velocities experienced in severe tornadoes and are hence vulnerable to wind damage as well as missile damage."

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* Philadelphia Electric Company, Limerick Generating Station, Final Safety Analysis Report.

The purpose of the Report is to provide a probabilistic assessment to demonstrate that the probability of exceeding 10 CFR Part 100 limits owing to wind effects on the ultimate heat sink (UHS) shall be less than or equal to a mean value of 10^{-6} per year, based on a conservative analysis.

2. APPROACH USED IN THE REPORT

As mentioned in Section 1 herein, the ultimate heat sink at Limerick consists of a spray pond which has four spray networks, each network having a capacity for shutdown of one unit. Failure of a spray network is defined as failure of the network itself or of the attendant feeder pipe. Thus, loss of the capacity of the UHS for shutdown of one unit occurs if all four spray networks have failed. Loss of the capacity of the UHS for shutdown of two units occurs if three of the four spray networks have failed.

We denote the probability of the event that three spray networks have failed by P(S). One approach to answering the NRC concern with respect to loss of the ESW and RHRSW functions and the consequent possible exceedance of 10 CFR 100 limits is to estimate by a conservative analysis the mean value of P(S), denoted by $\overline{P(S)}$. According to NRC criteria, it would be satisfactory if $\overline{P(S)} < 10^{-6}$ per year, regardless of whether one or two units were be in operation.

An alternative approach is to estimate by a "realistic" analysis the median of P(S), denoted by $\breve{P}(S)$. According to NRC criteria, it is satisfactory if $\breve{P}(S) < 10^{-7}$ per year, again regardless of whether one or two units were in operation.

Provided that $\overline{P}(S)<10^{-6}$ per year (or $\breve{P}(S)<10^{-7}$ per year), either of these approaches would tend to be conservative, since they do not account for the possibility that the capacity of the pond itself and/or of the cooling tower(s) would still be available after the occurrence of event S.

The Report does not use either of the approaches just described, and provides no estimates of $\overline{P(S)}$ or $\breve{P}(S)$. Rather, the Report uses an approach based on the observation that failure of four spray ponds (if one unit is operational) or of three spray ponds (if two units are operational) does not necessarily lead to the unavailability of the ESW and RHRSW systems (see last paragraph of quotation from Report in Section 1 herein). While the Report does not accord credit to the pond itself (as opposed to the spray networks), it does account for the capacity of the cooling towers to function as heat sinks unless damaged by strong winds or wind-borne missiles.

Thus, two damage states are defined in the Report:

"Damage state V: At least three out of four spray networks and both cooling towers are damaged. This is failure to provide a heat sink for the ESWS and RHRSWS when both units are operational.

Damage state T: All four spray networks and the Unit 1 cooling tower are damaged. This is failure to provide a heat sink for the ESWS and RHRSWS when only Unit 1 is operational."

The approach used in the Report consists of estimating the probabilities of occurrence $\overline{P}(V)$ and $\overline{P}(T)$ on the basis of an analysis described in the Report as being conservative. The estimates obtained in the Report are $\overline{P}(V)$ =

7.7×10^{-7} per year (when both units are operational), and $\overline{P}(T) = 6.6 \times 10^{-7}$ per year when only Unit 1 is operational.

3. EXTREME WIND ANALYSIS

The probability distribution estimated in the Report for tornado winds, hurricane winds, or winds other than tornadoes or hurricanes, is shown in Fig. 1. This distribution is acceptable from the point of view of its conservatism, as shown by a comparison of Fig. 1 with the probability distribution obtained by NRC and shown in Fig. 2. (See NRC memorandum from W. P. Gamill to O. D. Parr dated June 13, 1984.)

Once the curve of Fig. 1, which reflects tornadic as well as nontornadic winds, is established, the Report assumes that all winds that may occur on the site are associated with tornadoes. The Report further assumes that these tornadoes have such rates of occurrences as would result in the probability distribution curve of Fig. 1. The purpose of these assumptions is to allow the use of a computer program which is designed solely for tornado winds.

The Report provides no rationale for accepting the assumptions just described. Indeed, tornadic and nontornadic winds - which have vastly different flow structures - are not necessarily equivalent from the point of view of wind-borne missile effects if their respective speeds have the same probability of occurrence at a site. The difference between vertical mean speed profiles in tornadic and nontornadic winds also affects the estimates of the probability of failure of the cooling towers due to the direct aerodynamic action of nontornadic winds. Additional information concerning the effects of nontornadic winds was therefore requested form PECo. Such information was provided in reference 1 (p. 14) and reference 3. Reference 3 includes results of calculations on the effects on nontornadic winds based on hurricane and extratropical storm models which are reasonable both physically and climatologically.

4. SITE AND TARGET MODELING

The site model used in the computer simulations is based on a survey conducted in January 1984. The Report's assumptions relative to the missile origin zones (including structures that might fail and thus become sources of missiles), the relative distribution of missile types, the numbers of missiles available on the site, and the missile elevations, are all deemed to be reasonable.

For the sake of simplification, the spray nozzle networks are modeled as boxes extending horizontally 4 ft. and vertically 3 ft. beyond the volume totally enclosing the networks*. This extension accounts for the finite dimensions of the missiles.

To reduce the number of computer runs needed to simulate a sufficiently large number of missile hits, and to account for the finite dimensions of the missiles, the 30" diameter feeder pipes, which are partly exposed, were modeled as parallelepipeds with sides 33 ft. x 6 ft. The hit probabilities were then adjusted to account for that artificial increase. This procedure, too, is judged to be reasonable.

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* English units are used herein for the sake of consistency with the Report.

5. DAMAGE CRITERIA

5.1 SPRAY NETWORK

The Report assumes that a spray network fails completely if at least one missile of any type, j, hits the volume that models the network (see Sect. 4) with a sufficiently high speed, V_j^s.

The speed V_j^s is assumed to be the smaller of the speeds causing (a) the perforation of the thinnest distribution pipe in the network, and (b) the rupture of a spray arm.

As far as the speeds V_j^s corresponding to the perforation of the thinnest distribution pipe are concerned, the following observation is made in the Report. Since the motion of the missiles is predominantly horizontal, the chances that a missile will hit the surface obliquely are larger in the case of the horizontal top surface of the conceptual volume representing the spray network than in the case of the actual distribution pipes. To compensate for this difference, the thickness of the target surface is assumed to be equal to $1/\sqrt{2}$ times the thickness of the weakest distribution pipe.

Results of calculations concerning missile speeds that cause failure of the spray arms are given in Appendix C of the Report.

As indicated in pages 11 and 11a of the attachment to the memorandum from J. S. Kemper (PECo) to A. Schwencer (NRC) dated July 27, 1984, the direct aerodynamic effects of high winds upon the spray pond network pipes do not raise safety concerns.

5.2 FEEDER PIPES

Feeder pipes are subjected to the same adjustment for obliquity of missile hits as indicated above for the case of distribution pipes, i.e., a 0.354 in thickness is used in lieu of the 0.5 in nominal thickness of the pipes. Feeder pipes are also not affected adversely by the direct aerodynamic effects of high winds.

5.3 COOLING TOWERS

It is stated in the report that cooling towers are assumed to fail if the peak wind speed at the centerline of the tower at half height exceeds 140 mph, or if the distribution flume, the riser pipes, or the curb wall are damaged by wind-borne missiles. The assumption concerning the 140 mph was deemed to be in need of clarification and is further discussed in Sect. 5.4.2 herein.

5.4 COMMENTS ON EXTENT TO WHICH DAMAGE CRITERIA USED IN THE REPORT ARE CONSERVATIVE

5.4.1 Spray Networks

It was mentioned earlier that failure of a spray network is assumed to occur if at least one missile causes the postulated damage. It is the reviewer's opinion, based on experience with numerical simulations, that the occurrence of one missile hit with a large speed, V, is normally associated with the occurrence of a large number of hits by other missiles with speeds differing only insignificantly from V. Thus, the assumption just noted is likely to be only marginally, if at all, conservative.

Assumptions concerning the penetration of surfaces equivalent to pipes or the failure of spray nozzles hit by missiles are difficult to evaluate owing to the large number of uncertainties characterizing the pertinent phenomena. The reviewer believes that the assumptions used in the Report are reasonable. However, it is not justified to state that they are conservative given the aforementioned difficulties and uncertainties.

5.4.2 Cooling Towers

In order to estimate the probability of failure of the towers due   ne direct aerodynamic action of winds, it is necessary to convert the wind speeds to correspond to a 33 ft elevation. In view of the assumption used in the Report that the surrounding terrain is rough, there is a substantial reduction in the case of nontornadic winds. The Report does not deal with this satisfactorily. Clarifications concerning the behavior of the cooling towers were therefore requested from PECo and were provided in reference 1 (p. 12), reference 2 (under the heading Tower Failure), and reference 3 (p. 2). According to references 2 and 3, respectively, the calculated failure speeds for the towers are: 180 mph at 33 ft above ground for tornado winds (which have relatively slow variation of wind speed with height) and 135 mph at 33 ft above ground for hurricanes and extratropical storms (whose variation of wind speed with height is relatively strong, the roughness length being assumed to be $z_o \approx 3.3$ ft).

According to reference 2, these failure speeds are based on the assumption that Venturi (or interference) effects are not significant. This assumption may not be correct. This is suggested by Figs. 3 and 4, which show, for two cases, the ratios between tower stresses in the presence and in the absence of interference effects: these ratios may be as high as 15%, or even 30%. (In Figs. 3 and 4, min n_{22} = meridional compression, max n_{22} = meridional tension, max n_{11} = hoop tension, and so forth). Thus, to the extent that interference effects are neglected, the failure speeds would be overestimated.

On the other hand, in estimating the failure speed of the tower, the Report assumed that the pressures induced by the 2-sec wind gusts are perfectly correlated over the entire surface of the tower (see ref. 6). To this assumption there would correspond a gust response factor for the tower (i.e., a ratio between the response induced by the 2-sec wind gust and the response due to the mean hourly wind) of the order of 4.0. In reality, owing to the fact that spatial pressure correlations are imperfect, the gust response factor is significantly less than 4.0. (See Fig. 5.)

It thus appears that, in spite of possible Venturi (interference) effects, the 2-sec failure speeds at 33 ft above ground (180 mph for tornadic and 135 mph for nontornadic winds) given in ref. 2 and 3 are acceptable.

6. CONCLUSIONS

It was indicated in Sect. 3 herein that the estimates presented in the Report on tornado wind speeds are likely to be conservative. The other assumptions, pertaining to the behavior of the spray pond and of the cooling towers during tornado winds, are judged to be acceptable, but perhaps not conservative. Therefore, the conclusion presented in the Report to the effect that the mean probabilities of tornadoes causing the events of interest* are of the order of 10^{-6} or less per year, is acceptable.

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* That is, events V and T, defined in Sect. 2 herein.

The estimates presented in reference 3 show that the mean probabilities of nontornadic winds causing the events of interest are negligibly small compared to the probability of 10^{-6} per year. These estimates are based on the conservative assumption that the spatial correlations among pressures induced by 2-sec gust speeds are perfect over the entire surface of the tower.

It is therefore concluded that the probability of exceeding 10 CFR Part 100 limits owing to wind effects on the ultimate heat sink (UHS) and the cooling towers appears to be less than or equal to about 10^{-6} per year, and that this estimate appears to be marginally conservative.

REFERENCES

1. Letter from J. S. Kemper (PECo) to A. Schwencer (NRC) dated July 27, 1984 (Attachment).

2. Letter from J. S. Kemper (PECo) to A. Schwencer (NRC) dated Sept. 4, 1984 (Attachment).

3. Letter from J. S. Kemper (PECo) to A. Schwencer (NRC) dated Sept. 11, 1984 (Attachment).

4. H. J. Niemann, "Reliability of Current Design Methods for Wind-Induced Stresses," in Natural Draught Cooling Towers, Proceedings of the 2nd International Symposium, Ruhr-Universität Bochum, Germany, Sept. 5-7, 1984 (P. L. Gould et al., eds.), Springer-Verlag Berlin Heidelberg New York, Tokyo, 1984.

5. H. J. Niemann, "Wind Effects on Cooling-Tower Shells," Journal of the Structural Division, ASCE, Vol. 106, March 1980, pp. 643-661.

6. Letter from J. S. Kemper (PECo) to A. Schwencer (NRC) dated Sept. 24, 1984.

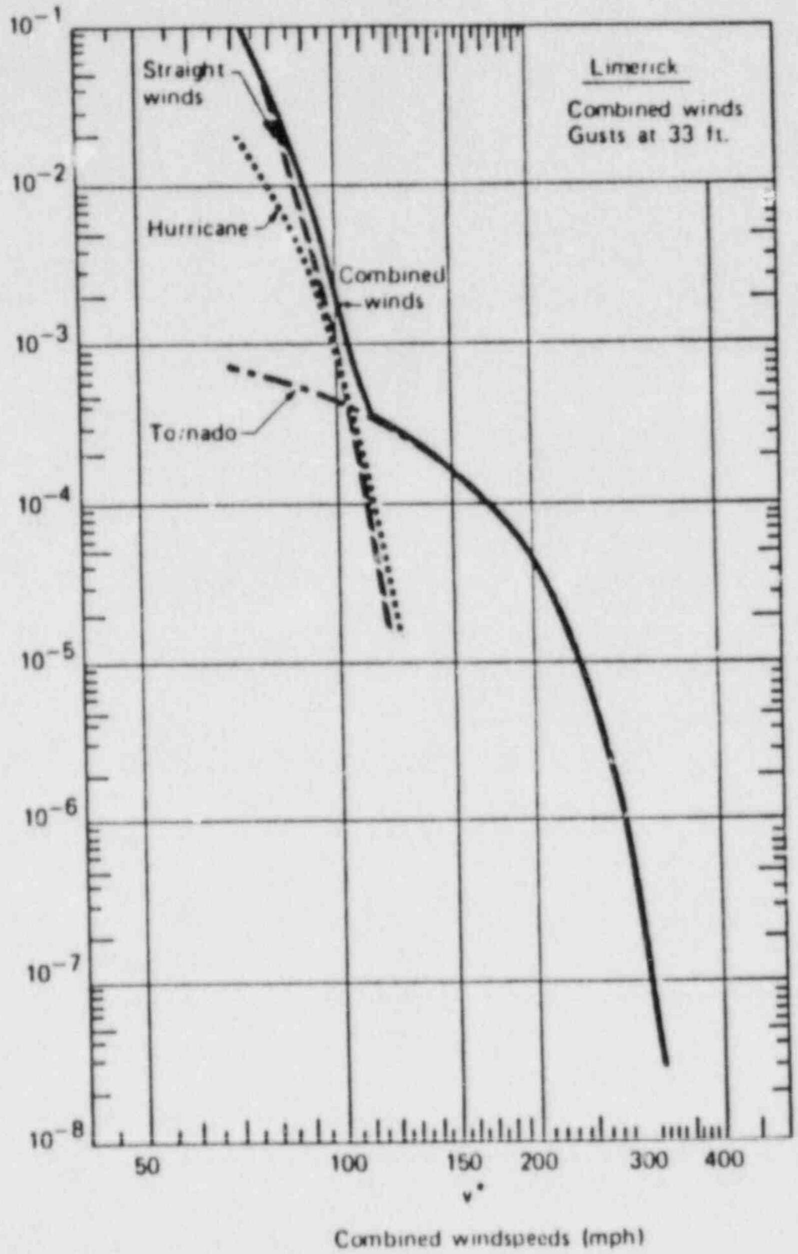


FIG. 1 - Extreme wind speeds (mph) as estimated in the Report.

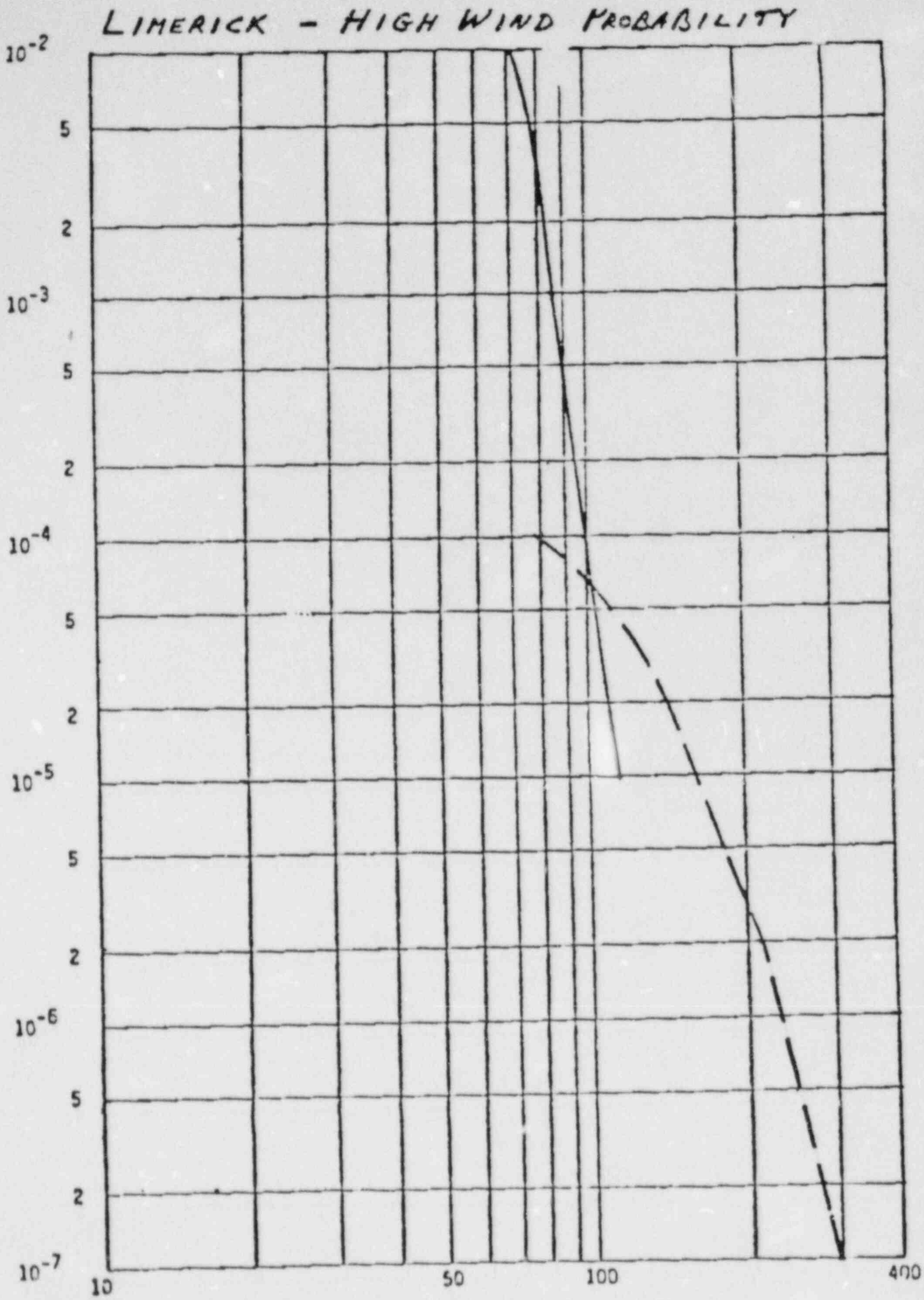
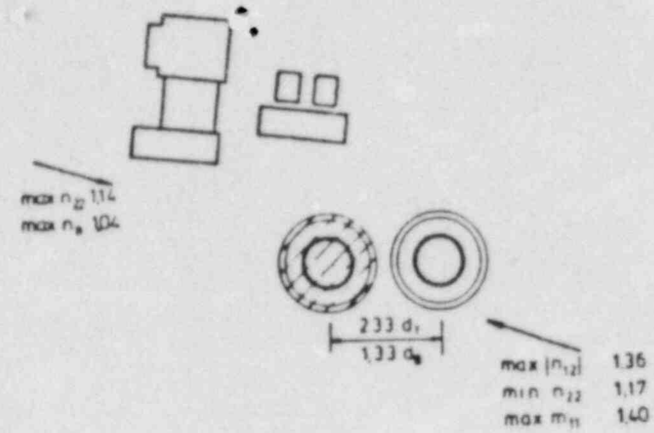
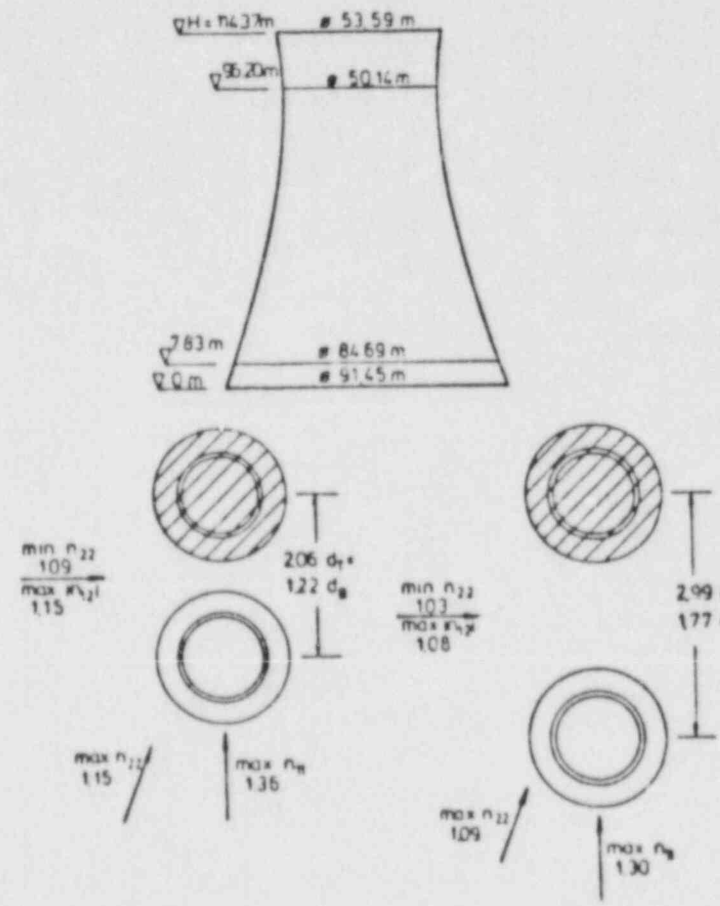


FIG. 2 - Extreme wind speeds (mph) as estimated by NRC.



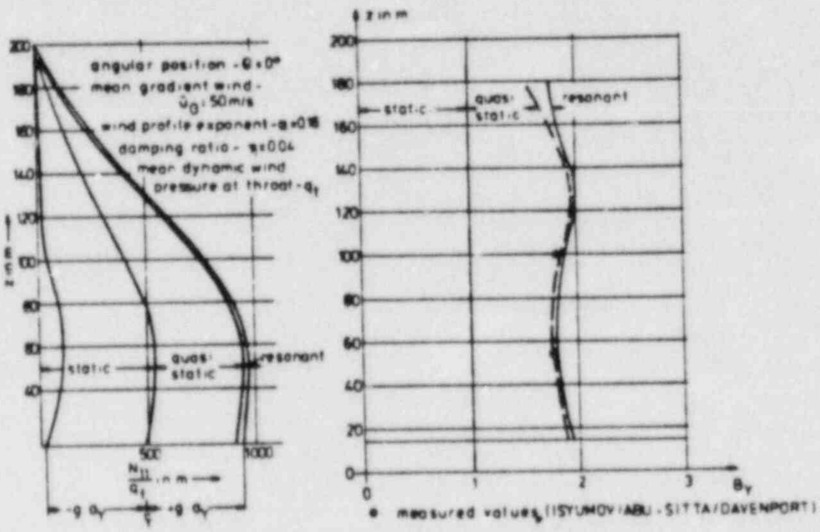
Interference factors in a power plant

FIG. 3 - (from Reference 4)



Increase of membrane forces in a group of two towers due to interference

FIG. 4 - (from Reference 4)



Meridional Stress Resultant N_{11} and Gust-Response Factor B_y along Meridian

FIG. 5 - (from Reference 5)

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

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In August 1983 the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0991) regarding the application of the Philadelphia Electric Company (the applicant) for licenses to operate the Limerick Generating Station, Units 1 and 2 located on a site in Montgomery and Chester Counties, Pennsylvania. Supplement No. 1 to NUREG-0991 was issued in December 1983 and addressed several outstanding issues. Supplement No. 1 also contains the comments made by the Advisory Committee on Reactor Safeguards in its report dated October 18, 1983. Supplement No. 2 was issued in October 1984 and addressed fourteen outstanding and fifty-three confirmatory issues and closed them out. This Supplement No. 3 to NUREG-0991 addresses the remaining issues that require resolution before issuance of the operating license for Unit 1 and closes them out.

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