NUREG-0991
Supplement No. 5

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

July 1985

NOTICE

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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 Counties, 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 interim report dated October 18, 1983. Supplement 2 was issued in October 1984. Supplement 3 was issued in October 1984 and addressed the remaining issues that required resolution before issuance of the operating license for Unit 1.

A license for the operation of Limerick Unit 1 was issued on October 26, 1984. The license, which was restricted to a five percent power level, contained conditions which required resolution prior to proceeding beyond the five percent power level. Supplement 4 to the SER, issued in May 1985, addressed some of the technical issues and their associated license conditions that required resolution prior to proceeding beyond the five percent power level. Supplement 4 to the SER also contained the comments made by the Advisory Committee on Reactor Safeguards in its report dated November 6, 1984, regarding full power operation of Limerick Unit 1. This Supplement 5 to the SER addresses further issues, that require resolution prior to proceeding beyond the five percent power level. As discussed in Section 13 of this report, a decision by the Commission authorizing full power operation of Unit 1 is dependant upon the progress of the Graterford prison emergency planning issues before the Atomic Safety and Licensing Board, the Atomic Safety and Licensing Appeal Board and the Commission, as appropriate.

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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 or the staff) issued its Safety Evaluation Report (NUREG-0991) regarding the application by the Philadelphia Electric Company (hereinafter referred to as the licensee) 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 and also contained the comments made by the Advisory Committee on Reactor Safeguards in its interim report dated October 18, 1983. In October 1984 Supplement 2 to NUREG-0991 was issued. Also in October 1984 Supplement 3 to NUREG-0991 was issued, addressing all remaining issues necessary to permit the issuance of an operating license. Operating License No. NPF-27, restricted to five percent power, was issued on October 26, 1984.

Supplement 4 to the SER, issued in May 1985, addressed some of the issues required to be resolved prior to proceeding beyond the five percent power level and also included the comments of the ACRS in its report dated November 6, 1984, regarding the full power operation of Limerick Unit 1. This Supplement 5 to the SER addresses further issues, that require resolution prior to proceeding beyond the five percent power level. As discussed in Section 13 of this report, a decision by the Commission authorizing full power operation of Unit 1 is dependant upon the progress of the Graterford prison emergency planning issues before the Atomic Safety and Licensing Board, the Atomic Safety and Licensing Appeal Board and the Commission, as appropriate.

The sections of this supplement are numbered the same as the corresponding section of the Safety Evaluation Report and Supplements No. 1 through 4. Each section is supplementary to and not in lieu of the discussion in the Safety Evaluation Report and Supplement No. 1 through 4 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 the 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 completion of the offsite emergency planning review was identified in SSER-3 as being relevant to the authorization of operations beyond the five percent power level. The results of the offsite emergency planning review, is reported in Section 13.3 of this report.

1.10 License Conditions

License No. NPF-27 contained conditions which require that certain actions be completed. Several of these actions must be completed prior to authorization to proceed beyond the five percent power level. The license conditions addressed by this supplement are as follows. The conditions are numbered as they were in previous SER Supplements.

License Conditions	Section
(6) Modifications to provide redundancy in remote shutdown system	7.4.2.3
(19) Emergency Response Capabilities	
(a) Detailed control room design review	18.1
(b) Safety Parameter Display System	18.2

1.12 Nuclear Waste Policy Act of 1982

Section 302(b) of the Nuclear Waste Policy Act of 1982 states that NRC shall not issue or renew a license for a nuclear power reactor unless the utility has signed a contract with the Department of Energy for disposal services. The Philadelphia Electric Company has signed a contractual agreement with the Department of Energy dated June 1, 1983. This agreement is applicable to Limerick Units 1 and 2.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.9 Mechanical Systems and Components

3.9.6 Inservice Testing of Pumps and Valves

In Section 3.9.6 of the SER and SSER 3, the staff stated that the relief the licensee requested from the pump and valve testing requirements of 10 CFR 50.55a(g) is warranted for that portion of the initial 120-month period during which the staff completes its detailed review. By a letter dated December 13, 1984, the licensee submitted Revision 5 of the Limerick Unit 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 5 for the same reasons as stated in the SER.

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 for Limerick Unit 1

This evaluation supplements conclusions in the corresponding section of Supplement No. 3 to the SER (SSER-3) which addresses the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).

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 Code Section XI requirements which it determined to be impractical to perform. These relief requests were supported by information pursuant to 10 CFR 50.55a(a)(3)*. The staff's evaluation of these submittals was reported in Appendix N of SSER 3. In a letter dated April 10, 1985, the licensee requested relief from additional Code requirements and provided a supporting technical justification. In the submittal, the licensee identified Relief Request Number 27 concerning support members for ASME Code Class 1 piping, Examination Category B-K-1. The licensee states that it has met the Code requirement for this examination category. However, in the licensee's Preservice Inspection Program a commitment was made to perform augmented examinations that exceed the requirements of the Code. The licensee was unable to complete the augmented examination on one support and submitted a request for relief from its Preservice Inspection Program. The staff has determined that the requirements defined in Section XI of the ASME Code provide an acceptable level of quality and safety. Since the requirements of the regulation were met the staff finds the licensee does not have to complete the commitment to exceed the requirements of the Code.

The staff evaluation consisted of reviewing the April 10, 1985 submittal and determining if relief from the Code requirements was justified. Pursuant to 10 CFR 50.55a(a)(3), relief from the Code requirements has been allowed for those requirements that, if implemented, would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. The evaluation of the additional relief requests is included in Appendix N to this report. On the bases of granting relief from these preservice examination requirements, the staff concludes that the Preservice Inspection Program for Unit 1 is in compliance with 10 CFR 50.55a(g)(2).

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*In the 1984 Edition of 10 CFR 50, this paragraph was designated as 50.55a(a)(2).

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.5 Combustible Gas Control

Inerting the containment for the LGS-1 plant is required by 10 CFR 50.44 (revised). In 10 CFR 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors," Section 50.44(c)(3)(i) states in part that, "Effective May 4, 1982 or 6 months after initial criticality, whichever is later, an inerted atmosphere shall be provided for each boiling light-water nuclear power reactor with a Mark I or Mark II type containment."

Evaluation

Since LGS-1 achieved its initial criticality on December 22, 1984, the plant is required to be inerted by June 22, 1985, per the 10 CFR 50.44 requirement set forth above. While the Limerick operating license NPF-27 has been limited to five percent of rated power the licensee has been unable to proceed with and complete the power ascension testing program (PATP) during the initial six months when the containment is not required to be inerted as originally planned. This has resulted in an extension of the time required to complete all post criticality PATP tests. The licensee requests the subject exemption to facilitate completion of the PATP. This requires that the licensee receive a temporary exemption from the requirement of 10 CFR 50.44 so that it may continue operating the plant with a non-inerted containment during the balance of the initial startup test program as originally planned.

The proposed exemption from the regulation is required in order to complete the balance of the PATP in accordance with the licensee's test plan. The licensee's test plan is based on maintaining the containment in a non-inerted condition until after completing the 100% rated thermal power trip test, a condition which normally would be expected to occur within about 120 effective full power days of core burn-up. No changes are being made in maximum full power days of core burn-up normally expected before inerting is required. In fact to assure this, the maximum expected value of 120 effective full power days is made part of the proposed action.

It is advantageous to operate the reactor without inerting during the PATP, as an uninerted containment would permit unscheduled inspections or identification of possible problems important to safety during this period. The anticipated high frequency of containment entries during the PATP period and the required deinerting and re-inerting time (about 24 hours) would tend to discourage early and frequent containment entries for identifying and correcting any potential safety problems before they become serious safety problems.

Further, the NRC staff believes that to now require inerting before the PATP tests have been completed could result in less assurance of safety, because of

the added time and/or decreased ability to directly examine and evaluate components and systems inside containment while the PATP tests are under way. Completing the PATP tests with an uninerted containment (exemption granted) then would reduce the likelihood of development of an event requiring protective safety actions both during the period of exemption and later. Because of the low level of fission product inventory during the PATP period, and the short duration anticipated for the exemption, there is an extremely low likelihood that the inerting system would be required.

Based on the information provided by the licensee and the staff's assurance that the remainder of the PATP tests will be performed in essentially the same manner as originally planned with respect to the magnitude and duration of power levels for each remaining PATP test, the NRC staff concludes that there will be no increase in the risks of operation through completion of the PATP tests with the proposed limited exemption regarding initial inerting over the risks that were contemplated for the duration of the PATP tests at the time the plant was licensed. Therefore, since there is no perceived increase in risk by the mere fact of extending the time allowed for completion of the PATP tests under uninerted conditions, the NRC staff finds that operation would be as safe under the conditions proposed by the exemption as it would have been had the test been completed in the shorter calendar time of six months after initial criticality.

After the containment has once been inerted, inspection personnel entering the containment after it has then been deinerted may be in some danger, because of the possibility that non-breathable nitrogen pockets may remain if the operator fails to initiate the mixing system. These risks are minimized during normal plant operation. However, during PATP, the risk is greater due to the large number of personnel entries into the containment.

The inerting requirement resulted from a staff judgement that the safety benefits attributable to having an inerted containment during normal operations outweighed the associated disadvantages. This judgement does not prevail during the PATP because of the need for frequent containment entries for inspection and surveillance purposes. The staff has reviewed the licensee's submittals, agrees with their statements and finds that the proposed exemption from 10 CFR 50.44(c)(3)(i) is acceptable.

With regard to the stage of the facility's life, LGS-1 construction is complete and the PATP is in progress through the five percent power level. Absent the requested exemption and consequent authorization to continue the PATP with a deinerted containment atmosphere, access to containment will be severely restricted. Frequent containment entries are required during PATP to adjust control systems, calibrate instruments and monitor containment conditions as the plant ascends in power. Without the requested exemption, considerable delay to deinert and reinert before and after containment entries will be encountered. At this point in the PATP, to require inerting would significantly extend the time to complete the PATP and, therefore, delay commercial operation.

The regulatory requirement from which the exemption is sought anticipated that power ascension test programs could be completed within six months and consequently the core fission product inventory that would build up over the life of the program was acceptable. While the regulation contemplated a six month

period, typical BWR programs have proven to actually require an average of 330 days. With this simple stretch in time, no significant increase in core inventory occurs and the same effective core history is experienced. Accordingly, for the reasons stated above, frequent containment entries, and the potential danger to the health and safety of plant operators, the staff finds that the containment should remain deinerted until completion of the PATP.

As discussed above, the staff concludes that in this instance an exemption from compliance with 10 CFR 50.44 for containment inerting has no adverse safety significance. Therefore, the granting of this exemption will have no effect on the public health and safety and will also promote efficient and expeditious testing of facility components and systems, and should therefore be granted.

Based on the considerations discussed above, we have concluded that the proposed temporary exemption from 10 CFR 50.44(c)(3)(i) is authorized by law, will not endanger life or property or the common defense and is otherwise in the public interest and should be granted.

6.6 Inservice Inspection of Class 2 and 3 Components

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

This evaluation supplements conclusions in the corresponding section of SSER-3 which addresses the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).

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 Code Section XI requirements which it determined to be impractical to perform. These relief requests were supported by information pursuant to 10 CFR 50.55a(a)(3)*. The staff's evaluation of these submittals was reported in Appendix N of SSER 3.

In a letter dated April 10, 1985, the licensee requested relief from additional Code requirements and provided a supporting technical justification. Therefore, the staff evaluation consisted of reviewing this submittal and determining if relief from the Code requirements was justified. Pursuant to 10 CFR 50, paragraph 50.55(a)(3), relief from the Code requirements has been allowed for those requirements that, if implemented, would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. The evaluation of the additional relief requests is included in Appendix N to this report. On the bases of granting relief from these preservice examination requirements, the staff concludes that the Preservice Inspection Program for Unit 1 is in compliance with 10 CFR 50.55a(g)(2).

*In the 1984 Edition of 10 CFR 50, this paragraph was designated as 50.55a(a)(2).

7 INSTRUMENTATION AND CONTROLS

7.4 Systems Required for Safe Shutdown

7.4.2 Specific Findings

7.4.2.3 Remote Shutdown System

In Supplement No. 3 to the SER (SSER-3) the NRC staff provided an evaluation of the status of the Limerick, Unit 1, remote shutdown capability and the basis for the granting of an interim exemption from GDC 19 for operation up to 5% power. A condition was included in License NPF-27 which required the licensee (1) to provide, prior to exceeding 5% power, 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 interim exemption was also based on the understanding that there was an existing single (primary) train of safety-related remote shutdown equipment.

The licensee responded to the low power license condition by letter dated February 15, 1985. The NRC staff requested additional information by letter dated March 6, 1985. The licensee responded by letters dated March 25 and April 18, 1985. The staff's evaluation of this information follows.

The interim redundant (back-up) remote shutdown system train of equipment has been fully implemented via the interim remote shutdown procedures. The interim redundant remote shutdown procedures require the use of temporary jumpers to establish control of the "B" RHR service water (SW), "B" RHR and "B" emergency service water (ESW) pumps, should the primary train of equipment fail to function. The staff has reviewed a drawing (Attachment 2 from the April 18, 1985 letter) which shows the installation of the temporary jumpers typical for the subject pumps' control circuits. Based on this review, we determined that the efforts involved in jumpering would be minimal. We therefore consider the use of jumpers for remote shutdown to be acceptable for the first cycle of plant operation.

To avoid jumpering in the final design, and to obtain a fully redundant safety-related train of remote shutdown equipment, the licensee has initiated design changes which will allow operation of the "B" RHRSW, "B" RHR, and "B" ESW pumps from their respective pump motor circuit breaker cubicles without the need to install temporary jumpers. A sketch illustrating the scope of the changes required was provided as Attachment 4 to the April 18, 1985 letter which shows that a keylocked, two-position, maintained contact transfer switch will be installed on each of the pumps' associated circuit breaker cubicles. The sketch is typical for all pumps. Upon transfer to the remote shutdown location, a control room annunciator will be activated identifying the associated equipment as out-of-service. The control room annunciator will not be able to be cleared unless the transfer switch is placed back into the control room operable position.

The licensee has verified that the transfer action will not cause a change in the operating status of the associated equipment. The transfer switch keys will be located in a seismically mounted box located in the remote shutdown room to enable ready access. The remote shutdown room is locked closed and access to the room is limited in accordance with the plant's existing administrative procedures. The final drawings required to implement the changes have not been issued. However, the licensee has provided, in Item I(c) of the April 18, 1985 letter, a commitment that the modifications will be engineered and implemented in accordance with safety-related design and qualification requirements, including seismic qualification. The pump motor control circuit changes are to be implemented during the first refueling outage. The staff finds the required modifications discussed above and schedule for implementation to be acceptable and requires that the licensee perform, prior to startup following the first refueling outage, design verification tests in accordance with the Technical Specification surveillance requirements for remote shutdown system instrumentation and controls. The objective of these tests is to demonstrate the operability of the modified system in accordance with GDC 19 remote shutdown system design requirements as interpreted by SRP Section 7.4. A condition in License NPF-39 has therefore been established to ensure that the required modifications and tests are satisfactorily completed. The NRC's Regional Office shall verify, as necessary, that this license condition is satisfied before plant restart after the first cycle of operation.

The condition in License NPF-27 also required the licensee 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 licensee provided a comparison of the existing (primary) single train of remote shutdown equipment with that of the interim redundant (back-up) remote shutdown train of equipment. Each primary component has been evaluated by the licensee for its worst case failure and each component's associated system design function has been verified to have been included in the interim redundant remote shutdown procedures. The licensee provided a detailed discussion of the interim procedures to be used should the existing (primary) remote shutdown train of equipment become inoperable. The steps for the installation of jumpers (discussed above), manual valve alignments, and use of the redundant (back-up) display instrumentation were reviewed. Some deficiencies were found related to the identification of what display instrumentation the operators should use if the primary train of equipment is not available. Upon the staff's request, the licensee upgraded the procedures to be more specific as to what instrumentation should be used and where it is located. The interim redundant remote shutdown procedures have been approved by the plant staff for implementation. The licensee has informed the staff that the approved interim procedures are currently in place in preparation for operation above 5% power. The NRC's Regional Office will verify, as necessary, the implementation of the interim redundant remote shutdown procedures.

The only design changes identified by the licensee required to obtain a fully redundant safety-related train of remote shutdown equipment were those associated with the installation of transfer switches as discussed above. The licensee has verified that the only impact this modification will have on the interim remote shutdown procedures will be the elimination of the steps required to install the temporary jumpers and the replacement of these steps with those referencing the operation of the transfer switches. The NRC's Regional Office

will verify, as necessary, the implementation of the final remote shutdown procedures prior to startup following the first refueling.

Some areas of concern resulted during the staff's review of this issue and warrants discussion. Review of the licensee's comparison between the existing (primary) train and back-up train of remote shutdown equipment resulted in the staff's discovery that various controls and instrumentation loops are not safety-related as the staff had understood from its review of the Limerick design prior to licensing. The licensee, upon the staff's request for justification, has upgraded several of the indication loops (RCIC flow, reactor vessel pressure, reactor vessel level, and RHR loop A flow) to comply with safety-related requirements including seismic qualification. Table 1 from the licensee's letter of April 22, 1985 lists the remote shutdown controls and indication and identifies what is or is not safety-related including seismically qualified. Furthermore, as requested by the staff, the licensee provided in the letter of April 22, 1985 information to justify why the remaining nonsafety-related controls and instrumentation do not need to be implemented as safety-related and, thus, are not needed for safe plant shutdown. The ICSB, in conjunction with RSB, reviewed the licensee's justification information and concluded that, should the non-safety-related equipment identified on the table not be available, the plant could be safety shut down and maintained in a safe shutdown condition from outside the control room using only redundant safety-related equipment in conjunction with the indicated jumpering for the first cycle of operation. The nonsafety-related equipment exists only for anticipatory purposes for which no credit is taken to perform the required remote safe shutdown function.

It should be noted that the staff's acceptance of various important indication loops (RHRSW pump discharge pressure, ESW pump discharge pressure, suppression pool level, and suppression pool temperature), even though exempt from safety-related requirements associated with safe plant shutdown at Limerick, was based on the implementation of appropriate currently existing Technical Specifications including requirements to ensure that the subject indication channels are periodically checked and calibrated to obtain some assurance of reliability should they be utilized for plant shutdown. Further, the acceptance of non-safety-related suppression pool temperature indication was based on the implementation of appropriate remote shutdown operating procedures related to the establishment of suppression pool cooling early in the event requiring control room evacuation. In a meeting held in Bethesda on April 22, 1985, the staff discussed with the licensee the pertinent portions of the remote shutdown procedures and Technical Specifications to verify the assertions being made.

Based on the foregoing, the staff finds that an exemption from full compliance with GDC 19 is justified for the period prior to startup following the first refueling outage. The staff also concludes that the requirements of Part A to Condition 2.c(14) in License NPF-27 have been satisfied and that this part of the condition is no longer necessary. The remaining requirements related to this issue are reflected in a conditon to the full power license. The information provided demonstrates that the interim remote shutdown procedure (to be verified as in place by Region I) and associated train of equipment in conjunction with the existing single (primary) remote shutdown train of equipment provides sufficient redundant remote shutdown capability for the first cycle of operation. With the satisfactory implementation of the identified design changes (transfer switches) at the first refueling outage and satisfactory completion of testing that will be required by the full power license condition,

the staff concludes that the final design will fully conform to the staff's acceptance criteria related to compliance with the requirements of GDC 19 as set forth in SRP Section 7.4.

11 RADIOACTIVE WASTE MANAGEMENT

11.4 Solid Waste Management System

In Section 11.4 of SSER 3, the staff stated that it found that the process control program (PCP) is acceptable pending acceptance of the final PCP to be submitted after analysis of the results of the preoperational testing to validate the process controls. In its March 7, 1985 letter transmitting its revised PCP, the licensee stated that it had completed preoperational testing of the solid radwaste system and had verified the acceptability of the process parameters. Changes that have been made to the PCP reflect information gained by the testing.

The staff has reviewed the March 7, 1985 submittal and has found the revised PCP to be acceptable with respect to its use for processing solid waste accumulations at the plant. This approval is granted on an interim basis pending the results of generic work by the staff. The generic effort results are expected to be used in independent reviews of PCP's for all operating plants with respect to compliance with 10 CFR Part 61.

13 CONDUCT OF OPERATIONS

13.3 Emergency Planning - Offsite

13.3.1 Federal Emergency Management Agency (FEMA) Findings on Offsite Emergency Plans and Preparedness

FEMA has completed its review of the offsite radiological emergency plans and preparedness in support of licensing for the Limerick Generating Station and has provided its findings and determinations to the NRC. There are three counties and 42 municipalities within the plume exposure pathway Emergency Planning Zone for Limerick. On May 8, 1984, FEMA provided its initial interim finding on the offsite plans and identified four "Category A" deficiencies; i.e., deficiencies that would lead to a negative finding, as well as a number of lesser deficiencies. On September 25, 1984, FEMA provided its report for the July 25, 1984 joint full participation exercise of the offsite plans. In this report FEMA identified five Category A deficiencies three of which were related to deficiencies identified in the earlier plan review.

A supplemental exercise of the offsite plans was conducted on November 20, 1984 to address the Category A deficiency related to lack of participation by some of the offsite jurisdictions identified in the July 25, 1984 exercise. In a report to the NRC dated January 10, 1985, FEMA indicated that the supplemental exercise resolved the deficiency, however, a Category A deficiency related to the non-participation of one municipality was identified by FEMA in the exercise. On March 27, 1985, FEMA informed the NRC that a remedial exercise conducted at the Graterford State Correctional Institution on March 7, 1985 demonstrated that a portion of one of the Category A deficiencies identified in the July 25, 1984 exercise had been corrected.

On April 9, 1985, FEMA provided a supplemental interim finding on the status of the ongoing review of the revised offsite response plans for Limerick which identified six remaining plan and preparedness deficiencies. In a memorandum dated May 21, 1985, FEMA transmitted exercise evaluation reports for remedial exercises conducted on April 10 and 22, 1985 and provided its final interim finding on offsite planning and preparedness. FEMA stated that as a result of the July 25, 1984 full participation exercise, the November 20, 1984 supplemental exercise, the March 7, April 10 and April 22, 1985 remedial exercises, all Category A deficiencies requiring demonstration have been corrected. In addition, FEMA reported that all Category A planning deficiencies have also been corrected. FEMA concluded that offsite radiological emergency planning and preparedness is now adequate to provide reasonable assurance that protective measures can be implemented to protect the public health and safety in the event of a radiological emergency at the Limerick Generating Station.

13.3.2 Offsite Emergency Planning Medical Services

In a recent decision, GUARD v. NRC, 753 F.2d 1144 (D.C. Cir. 1985), the U.S. Court of Appeals vacated the Commission's interpretation of 10 CFR §50.47(b)(12)

to the extent that a list of facilities was found to constitute adequate arrangements for medical services for members of the public offsite exposed to dangerous levels of radiation. The Commission has now provided guidance to be followed in determining compliance with this regulation pending its determination of how it will proceed in response to the Court's remand. In particular, the Commission directed that Licensing Boards, and in uncontested cases, the staff, should consider the uncertainty attendant to the Commission's interpretation of this regulation, especially in regard to its interpretation of the term "contaminated injured individuals." IN GUARD, the Court left open to the Commission the discretion to reconsider whether that term should include members of the offsite public exposed to dangerous levels of radiation and, thus, whether arrangements for this population of individuals are required at all. For this reason, the Commission observed that it may reasonably be concluded that "no additional actions should be taken now on the strength of the present interpretation of that term." Accordingly, the Commission observed that it can be found "that any deficiency which may be found in complying with a finalized post GUARD planning standard (b)(12) is insignificant for the purposes of 10 CFR §50.47(c)(1)." In this regard, the Commission, as a generic matter, noted the low probability of accidents which might result in exposure of members of the offsite public to dangerous levels of radiation as well as the slow development of adverse reactions to overexposure. See, Emergency Planning; Statement of Policy, 50 FR 20892, May 21, 1985.

Consistent with the foregoing Statement of Policy, the applicant has, by letter dated May 31, 1985, confirmed that, in good faith reliance on the Commission's earlier interpretation of 10 CFR §50.47(b)(12), the emergency plans of the involved offsite response jurisdictions contain a list of medical service facilities. The existence of such list in the pertinent plans has also been confirmed by FEMA. As stated by the Commission, such good faith reliance, in the circumstances, can be found to constitute "other compelling reasons" within the meaning of 10 CFR §50.47(c)(1). Further, the applicant has committed to fully comply with the Commission's response to the Court's remand.

Accordingly, on the basis of the factors identified by the Commission in its Statement of Policy, the staff has determined that the requirements of 10 CFR §50.47(c)(1) have been satisfied so as to warrant issuance of the operating license pending further action by the Commission with respect to the requirements of 10 CFR §50.47(b)(12).

13.3.3 Atomic Safety and Licensing Board Conditions

On May 2, 1985, the Atomic Safety and Licensing Board issued its Third Partial Initial Decision on Offsite Emergency Planning for the Limerick Generating Station. The Board found in favor of the applicant on all issues in contention, except for any issue which may arise regarding the inmates of Graterford prison. The Board imposed the following two conditions that are to be satisfied prior to operation above 5% of rated power:

(1) The Director, Office of Nuclear Reactor Regulation shall receive verification of plans to implement a level of traffic control in the King of Prussia area sufficient to assure that all the traffic evacuating along the Route 363-to-Pennsylvania Turnpike can continue to move upon reaching the EPZ boundary.

(2) FEMA shall receive verification of the satisfaction of the unmet municipal staffing needs as they relate to a capability of continuous 24-hour operation during a radiological emergency.

Since both conditions are related to offsite matters, the staff requested the assistance of FEMA in responding to the Board conditions. FEMA responded in the May 21, 1985 memorandum and in a May 30, 1985 update. Regarding the first condition, FEMA stated that they have received verification that traffic and access control points will be established and manned to ensure that evacuating traffic could continue moving upon reaching the Emergency Planning Zone (EPZ) boundary, and that this information will be incorporated into the Montgomery and Chester County plans. Regarding the second condition, FEMA stated that they have determined that adequate staffing now exists in all risk municipalities to respond to a radiological emergency over an extended period of time. Based on the information provided by FEMA, the staff concludes that the two Board conditions have been satisfactorily resolved.

13.3.4 Atomic Safety and Licensing Board Decision Related To Graterford Prison

The Atomic Safety and Licensing Board issued an order on June 12, 1985, admitting two of the contentions that were submitted by the inmates at the State Correctional Institute at Graterford, Pennsylvania on May 13, 1985. These contentions concern training for civilian emergency response personnel such as bus and ambulance drivers and the development of evacuation time estimates for the prison facility which is located within the plume exposure pathway Emergency Planning Zone approximately 8.3 miles east of the Limerick Station. A hearing on these contentions is scheduled to begin on July 15, 1985.

13.3.5 Conclusion on Emergency Preparedness

Based on a review of the FEMA findings and determinations on the adequacy of offsite radiological emergency response plans and preparedness, and on the previous NRC assessment of the adequacy of the onsite emergency plans and preparedness (see SSER No. 3 dated October 1984), the NRC staff concludes that, with respect to these issues, the overall state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the Limerick Generating Station. A decision by the Commission authorizing full power operation of Unit 1 is also dependant on the progress of the Graterford prison issues as discussed above before the Atomic Safety and Licensing Board, the Atomic Safety and Licensing Appeal Board and the Commission, as appropriate.

14 INITIAL TEST PROGRAM

One of the bases for issuing an operating license is that a plant will be tested in accordance with a staff-approved initial test program. Frequently, however, it is desirable for licensees to modify the approved test program due to the temporary unavailability of certain equipment or other factors. The purpose of the related condition in license NPF-39 is to ensure that any safety significant deviations from the staff-approved test program are identified to the staff in a timely manner. Experience has shown that some applicants/licensees have not attributed the appropriate level of importance to safety of some structures, systems and components. Therefore, this license condition requires that 10 CFR 50.59 changes to the test program be reported within one month to provide added assurance that the plant is not operated for an extended period of time at power levels at which its safety is dependent on untested or inadequately tested structures, systems or components. Furthermore, the one month reporting period provides reasonable assurance of NRC awareness of most test program changes prior to completion of the test program.

15 ACCIDENT ANALYSIS

15.8 Anticipated Transients Without Scram Generic Letter 83-28, Item 1.1 - Post-Trip Review (Program Description and Procedure)

Introduction

On February 25, 1983, both of the scram circuit breakers at Unit 1 of the Salem Nuclear Power Plant failed to open upon an automatic reactor trip signal from the reactor protection system. This incident occurred during the plant start-up and the reactor was tripped manually by the operator about 30 seconds after the initiation of the automatic trip signal. The failure of the circuit breakers has been determined to be related to the sticking of the under voltage trip attachment. Prior to this incident, on February 22, 1983, at Unit 1 of the Salem Nuclear Power Plant, an automatic trip signal was generated based on steam generator low-low level during plant start-up. In this case, the reactor was tripped manually by the operator almost coincidentally with the automatic trip. Following these incidents, on February 28, 1983, the NRC Executive Director for Operations (EDO) directed the staff to investigate and report on the generic implications of these occurrences at Unit 1 of the Salem Nuclear Power Plant. The results of the staff's inquiry into the generic implications of the Salem unit incidents are reported in NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant." As a result of this investigation, the Commission (NRC) requested (by Generic Letter 83-28 dated July 8, 1983) all licensees of operating reactors, applicants for an operating license, and holders of construction permits to respond to certain generic concerns. These concerns are categorized into four areas: (1) Post-Trip Review, (2) Equipment Classification and Vendor Interface, (3) Post-Maintenance Testing, and (4) Reactor Trip System Reliability Improvements.

The first action item, Post-Trip Review, consists of Action Item 1.1, "Program Description and Procedure" and Action Item 1.2, "Data and Information Capability." This safety evaluation report (SER) addresses Action Item 1.1 only.

REVIEW GUIDELINES

The following review guidelines were developed after initial evaluation of the various utility responses to Item 1.1 of Generic Letter 83-28 and incorporate the best features of these submittals. As such, these review guidelines in effect represent a "good practices" approach to post-trip review. We have reviewed the applicant's response to Item 1.1 against these guidelines:

A. The licensee or applicant should have systematic safety assessment procedures established that will ensure that the following restart criteria are met before restart is authorized.

   - The post-trip review team has determined the root cause and sequence of events resulting in the plant trip.

- Near term corrective actions have been taken to remedy the cause of the trip.

- The post-trip review team has performed an analysis and determined that the major safety systems responded to the event within specified limits of the primary system parameters.

- The post-trip review has not resulted in the discovery of a potential safety concern (e.g., the root cause of the event occurs with a frequency significantly larger than expected).

- If any of the above restart criteria are not met, then an independent assessment of the event is performed by the Plant Operations Review Committee (PORC), or another designated group with similar authority and experience.

B. The responsibilities and authorities of the personnel who will perform the review and analysis should be well defined.

- The post-trip review team leader should be a member of plant management at the shift supervisor level or above and should hold or should have held an SRO license on the plant. The team leader should be charged with overall responsibility for directing the post-trip review, including data gathering and data assessment and he/she should have the necessary authority to obtain all personnel and data needed for the post-trip review.

- A second person on the review team should be an STA or should hold a relevant engineering degree with special transient analysis training.

- The team leader and the STA (Engineer) should be responsible to concur on a decision/recommendation to restart the plant. A nonconcurrence from either of these persons should be sufficient to prevent restart until the trip has been reviewed by the PORC or equivalent organization.

C. The licensee or applicant should indicate that the plant response to the trip event will be evaluated and a determination made as to whether the plant response was within acceptable limits. The evaluation should include:

- A verification of the proper operation of plant systems and equipment by comparison of the pertinent data obtained during the post-trip review to the applicable data provided in the FSAR.

- An analysis of the sequence of events to verify the proper functioning of safety related and other important equipment. Where possible, comparisons with previous similar events should be made.

D. The licensee or applicant should have procedures to ensure that all physical evidence necessary for an independent assessment is preserved.

E. Each licensee or applicant should provide in its submittal, copies of the plant procedures which contain the information required in Items A through D. As a minimum, these should include the following:

- The criteria for determining the acceptability of restart
- The qualifications, responsibilities and authorities of key personnel involved in the post-trip review process
- The methods and criteria for determining whether the plant variables and system responses were within the limits as described in the FSAR
- The criteria for determining the need for an independent review.

EVALUATION AND CONCLUSION

By letter dated August 31, 1984, the licensee for the Limerick Generating Station, Units 1 and 2, provided information regarding its post-trip review program and procedures. The NRC staff has evaluated the licensee's program and procedures against the review guidelines developed as described in Section II. A brief description of the licensee's response and the staff's evaluation of the response against each of the review guidelines is provided below:

A. The licensee has established the criteria for determining the acceptability of restart. Based on our review, we find that the applicant's criteria conform to the guidelines as described in the above Section II.A, and, therefore, are acceptable.

B. The qualifications, responsibilities and authorities of the personnel who will perform the review and analysis have been clearly described. We have reviewed the licensee's chain of command for responsibility for post-trip review and evaluation and find it acceptable.

C. The licensee has described the methods and criteria for comparing the event information with known or expected plant behavior. Based on our review, we find them to be acceptable.

D. The licensee has established criteria for determining the need for independent assessment of an event. Based on our review, we find them acceptable. In addition, the licensee has established procedures to ensure that all physical evidence necessary for an independent assessment is preserved. We find that this action to be taken by the licensee conforms with the guidelines as described in the above Sections II.A and D.

E. The licensee has provided for our review a systematic safety assessment program to assess unscheduled reactor trips. Based on our review, we find that this program is acceptable.

The NRC staff concludes, based on its review, that the licensee's Post-Trip Review Program and Procedures for Limerick Generating Station, Units 1 and 2, are acceptable.

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 licensee submitted the Limerick Summary Report on August 16, 1984 (Ref. 1) and the staff's evaluation of the DCRDR had been reported on in Supplement No. 3 to the Safety Evaluation Report (SSER-3). The results of our SSER-3 evaluation concluded that the Limerick Unit 1 DCRDR was meeting the NUREG-0737, Supplement 1 requirements for work completed, but that the DCRDR was incomplete. The incomplete portion of the DCRDR consisted of:

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

To ensure the timely and the satisfactory completion of the DCRDR, the staff included license conditions on the above issues in the Limerick Generating Station, Unit 1 Facility Operating License No. NPF-27. These conditions are restated in the appropriate sections below.

18.1.2 Evaluation of Detailed Control Room Design Review

In November 1984, the licensee submitted Supplement 1 to the DCRDR Summary Report (Ref. 2) for the staff's review. This supplement documents the completion of the final validation step of the Limerick DCRDR Program Plan.

Supplement 1 to the Summary Report documents the results of DCRDR activities that have been completed by Limerick subsequent to the submittal of the Summary Report. These activities are stated as DCRDR Validation, disposition/resolution of outstanding Control Room Survey (CRS) items, implementation of actual control room enhancements, and results of a follow-on meeting among the NRC, Lawrence Livermore National Laboratory (LLNL), Philadelphia Electric Company (PECO), and The Interlock Group, the human-factors consultants to PECO. However, these activities do not complete Limerick's DCRDR.

A major item of work in Limerick's DCRDR is the use of function and task analysis to identify control room operator tasks and information and control requirements during emergency operations. In addition, a comparison of display and control requirements with a control room inventory to identify missing displays must be performed. In the DCRDR Summary Report, the licensee requested a delay in the completion of the task analysis until June 1985. The staff's review and acceptance of the request is reported in SSER-3 and is reflected in condition 2.C(8)(a) Part 1 to License NPF-27. By letter dated June 28, 1985, the licensee submitted CRDR Final Report Supplement No. 2, which addresses these issues, thus satisfying the requirements of license condition 2.C(8)(a) Part 1.

The NRC staff's safety evaluation of Limerick's DCRDR activities described in Supplement 1 to the DCRDR Summary Report is presented in this Supplement No. 5 to the SER. The evaluation results have been structured in the same format used to present the initial safety evaluation.

18.1.3 Review Team and Review Program

The NRC staff review of this issue has been completed and closed out in SSER-3.

18.1.4 System Function and Task Analysis

As noted above, the licensee's response to the requirement to submit the results from the Limerick plant specific task and information/control requirements for emergency operations, including a complete description of the method, data, and documentation, has been reported to the staff by letter dated June 28, 1985, pursuant to condition 2.C(8)(a) Part 1 in License NPF-27. The licensee states that this effort also includes a comparison of the display and control requirements with a control room inventory to identify missing displays and controls.

18.1.5 Control Room Inventory

The staff's concerns on this issue are expressed in Section 18.1.4 above and also in SSER-3. The licensee's letter of June 28, 1985, states that actions in response to this item have also been completed pursuant to condition 2.C(8)(a) Part 1 of License NPF-27.

18.1.6 Control Room Survey

In SSER-3 the staff stated that several elements in the control room survey were incomplete because control room construction was incomplete. The incomplete elements of the survey were:

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

In addition, SSER-3 also identified several human engineering deficiencies (HEDs) which were found by the NRC audit team during our in-progress audit. Based on this data, the staff established a condition to operating License NPF-27 to

ensure completion of the survey and the correction of high safety significance HEDs which may result from the survey:

> Condition 2.C(8)(a), Part 2, Control Room Survey
>
> The licensee shall satisfactorily complete the control room survey, evaluate all human engineering discrepancies defined by the survey, including those defined by the staff's audit team during the In-Progress Audit, and correct human engineering discrepancies which have been categorized as Priority 1 (High Safety Significance) prior to operation at a power level greater than five percent of rated power. The results from the effort are to be documented in an addendum to the Final Report and submitted for staff review.

In Supplement 1 to the Summary Report (the report), the licensee states that the Control Room Survey effort is completed and was accomplished using the Boiling Water Reactors Owners' Group (BWROG) Control Room Survey checklists (original and supplemental). Section 1.3 of Supplement 1 to the Summary Report (Ref. 2) lists the checklists that were administered.

The report states that seven computer related human engineering discrepancies (HEDs) were discovered. These HEDs were for keyboard design, terminology and visibility items, including software terminology. In addition to the computer HEDs, two procedure-related HEDs were defined. These HEDs were concerned with component identification and related procedure referencing. Further, the report states that the survey of control room environment, maintenance and surveillance, and training and manning revealed no discrepancies.

The staff evaluated the survey activities described and the results stated in Supplement 1 to the licensee's Summary Report. The activities described and the results stated appear to be consistent with other DCRDR reviews conducted by the staff. However, the staff noted that the licensee did not assess all of the HEDs found by the NRC audit team during the In-Progress Audit. These specific HEDs are stated under the title of Process Computers, Part A, Appendix A of our SER. For confirmatory review, the staff requested the licensee to assess and to report to the staff the results of the assessment of all of the HEDs identified by the NRC audit team.

In response the licensee states in an April 4, 1985 letter (Ref. 4) that all human engineering discrepancies (HEDs) identified during the NRC's in-progress audit have been integrated into the review process. At this point in the review, the surveys have been completed, with the exception of a survey of the computer based Safety Parameter Display System (SPDS).

In an April 12, 1985 letter (Ref. 6), the licensee states that the SPDS is not operational at this time but commits to perform a human-factors survey of the SPDS prior to its operational use by control room operators. The licensee provides the following reasons why this delay is acceptable:

"1. The SPDS itself will not be operable until its validation and testing program is complete.

2. General Electric, the system designer of the SPDS, has hired a human factors consultant to assist in the man-machine interface development.

The NRC has performed a Design Verification Audit of the General Electric design effort and reported the results of their review in Supplement No. 3 of the Limerick SER. (Section 18.2.5)

3. During the Limerick CRDR validation effort a hardcopy of the SPDS displays were used to verify their coordination with the CRDR."

Based on the NRC staff's review of these reasons and the licensee's commitment to perform a human-factors survey of the SPDS, the staff finds the delay of this one survey acceptable. However, the staff recommends that the licensee give special attention to the text, labels, and units used within the SPDS display formats in comparison to the standards adopted for the Limerick control room.

On the basis of the three reasons discussed above, the licensee concludes that there is a very low probability that the results from the SPDS survey to be conducted in the future will identify any Priority 1 or Priority 2 HEDs. The staff agrees that the probability of identification of further Category I HEDs due to the SPDS surveys is low and concludes that the licensee's commitment to perform a survey of the SPDS prior to its operation is sufficient to meet the requirements of Supplement 1 to NUREG-0737 in this regard. Therefore, based on the above, the staff concludes that the licensee's response to condition 2.C(8)(a) Part 2 is adequate and this portion of the condition is no longer necessary.

18.1.7 Assessment of HEDs

Condition on Control Room Enhancements

Based on the results of the staff's previous review of the licensee's progress in the assessment and correction of HEDs, the staff established a condition to operating license NPF-27 to ensure timely completion of control room enhancements:

Condition 2.C(8)(a), Part 3, Control Room Enhancements

The licensee shall complete control room enhancements related to: the control room panels (paint, tape and label), re-scaling of instruments using acceptable human-factors methods, and changes to standard control switch shapes prior to exceeding five percent of rated power.

Subsequently, the licensee in Supplement 1 to the Summary Report states that the panel enhancement effort was completed in mid-September 1984. Photographs of several control room panels before and after enhancement were provided in the report. Further, the November 1984 report stated that refinements to the enhancements were continuing. The licensee now states in a March 21, 1985 letter (Ref. 3) as supplemented by letters dated May 24, 1985 and June 10, 1985, that these enhancements, consisting of paint, tape and label improvements to panels, rescaling of instruments using human factors methods, and changes to standard control switch shapes, are completed. Based on this information, the staff concludes that the licensee's response to condition 2.C.(8)(a) Part 3 is adequate and this portion of the condition is no longer necessary.

NRC Audit Team HEDs

In December 1983, an NRC Audit Team conducted an in-progress audit of the Limerick DCRDR and identified 17 HEDs. The licensee provided data (Ref. 4) which indicated that nine of these HEDs had been previously identified and resolved by the Limerick Review Team. The eight remaining HEDs were to be assessed for safety significance and the results reported to the staff. The licensee indicated that a preliminary assessment of these HEDs had failed to identify any Priority 1 (High Safety Significance) HEDs.

In Reference 6, the licensee provided the staff with details on each of the eight remaining HEDs. In addition, it states that the assessment of these HEDs resulted in no Priority 1 (High Safety Significance) HEDs and only one Priority 2 (Low Safety Significance) HED, which has subsequently been corrected. Based on the review of this material, the staff now concludes that all of the HEDs identified by the NRC Audit Team during the in-progress audit have been integrated into the review process.

HED SI4-04

One high priority HED reviewed by the staff is HED SI4-04. This HED states that failed indicator light bulbs on the remote shutdown panel cannot be distinguished from a normal condition. The licensee planned actions as a temporary resolution to this HED. In the Summary Report, the licensee stated a commitment to complete these planned actions prior to operation of the plant at a power level greater than 5 percent of rated power. The licensee states in a letter dated April 4, 1985 (Ref. 4) that the correction of HED SI4-04 which defined a potential monitoring problem with failed indicator light bulbs on the remote shutdown panel that could not be distinguished from a normal condition has been completed and implemented. Based on this information the staff's concerns on HED SI4-04 are resolved

HED SPV-07

Supplement 1 to the licensee's report discussed a total of 36 HEDs which were found during the DCRDR activities. The sources of the HEDs were as follows:

Control Room Validation	
- Panel design	15
- Instrumentation	5
- Procedural	7
Computer	7
Procedures	2
Total	36

The report states that none of these HEDs were assessed as Priority 1 (Safety Significant). Of the 36 HEDs, 15 were assessed as Priority 2 (Low Safety Significance), 7 were Priority 4 (no significant improvement), and 14 were assessed as not being discrepancies. HED Assessment Forms on all 36 HEDs were included in the report.

The staff reviewed the contents of the 36 HED Assessment Forms and concluded that most were correctly categorized. The staff discussed the categorization of these HEDs with the NRC Resident Inspector for Limerick and examined in

detail HED SPV-07, which deals with inconsistencies between panel ID numbers/ nomenclature and procedure valve numbers/nomenclature within the T-200 series of procedures.

This HED was assessed as Priority 2, with resolution scheduled by the first refueling. The resolution for the HED was to make all procedure nomenclature and identification numbering consistent with CRDR assigned nomenclature and identification numbers.

In the staff's assessment of the T-200 Series Procedures, it was noted that many of these procedures were for the operation of the Engineered Safety Feature Systems. Engineered Safety Feature systems must be operated correctly to successfully mitigate the consequences of an accident. As the procedure nomenclature is inconsistent with updated panel nomenclature, operator errors may be made in the execution of the procedures. The staff requested the licensee to reassess HED SPV-07 and to provide further justification for the priority and resolution schedule of this HED.

In response the licensee stated in a letter dated April 12, 1985 (Ref. 6) that:

> "The correction of HED SPV-07 (T-200 Series Procedures) has been completed. All T-200 series procedures have been revised and approved by the Plant Operations Review Committee (PORC) to properly reflect the enhanced hierarchical labeled control room."

Based on this information, the staff's concerns on HED SPV-07 are resolved.

Summary

The staff also assessed the number and type of HEDs defined by the licensee's DCRDR. To date, 225 HEDs have been defined by the licensee's effort (189 HEDs in the Summary Report, and 36 HEDs in Supplement 1 to that report. Twenty-seven of these HEDs were identified during validation activities. Of the 27 HEDs, 18 were function/system type HEDs typical of the type of HEDs anticipated from validation activities. In the staff's judgment, the 9 remaining HEDs could have been detected during earlier DCRDR verification activities. However, as the 9 HEDs represent only four percent of the total HEDs, it appears that the previous verification activities conducted by the licensee were effective. In addition, the 27 HEDs defined from validation activities represents 12 percent of the total HEDs. This low figure also substantiates the NRC staff's general conclusion on the high quality of previous verification activities.

At this point, the staff concludes that a large portion of the HED assessment effort within the DCRDR has been completed. The effort which remains pertains to the assessment of HEDs which may be identified as the result of the task analysis and control room inventory, and from the human-factors survey of the SPDS.

18.1.8 Selection of Design Improvements

The NRC staff's initial evaluation of this issue is reported in SSER-3. The staff will evaluate the effectiveness of the licensee's process used in the selection of design improvements and report the results in a future supplement to the SER.

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

The evaluation in SSER-3 of the methods proposed by the licensee to achieve the above objectives concluded that the intent of NUREG-0737, Supplement 1 was being met. In evaluating the licensee's Supplement 1 to the Summary Report, the staff assessed the licensee's effort and results from the implementation of the proposed methods. Figures 1 and 2, of the licensee's report show photographs of control room panels before and after enhancement.

The staff analyzed the photographs and noted that several portions of the workbench and panel area were enhanced with a red background. The intent is to color code the background for operator ease in the location and use of the enclosed instruments and controls. The staff also noted that HED SDV-10, Automatic Depressurization System (ADS) Valve Enhancements, was resolved by enclosing five ADS valve controls in solid red background. The staff is concerned that red status lights for the valves would not be detected within the solid red background areas of the workbench and panel. The NRC Resident Inspector at Limerick was requested to evaluate this problem. The NRC Resident Inspector communicated (by phone) the results of his assessment, that contrast existed between lighted, red status bulbs and the red background and the lighted status bulbs could be easily detected for normal and emergency lighting conditions. However, the Inspector noted that when the status bulb was unlighted, it blended with the solid red background. The Inspector pointed out that a potential loss of operator information may exist during a loss of electrical power, that is, a condition where the status bulb should be lighted, but is not, and this would not be noted by the operator as the bulb blends with the red background.

The staff requested the licensee to provide data verifying the use of a solid red background as an enhancement to areas of the workbench and panels. The licensee responded (Ref. 4) that a red status light against a red background is visible and detectable by operators. They also state that the absence of any lighted bulb identifies a failed light bulb. In addition, the red background was used during the final validation phase on the full scale mock-up of the control room and no problems were identified on this issue. Further, the control room enhancements were completed (October 1984) with this red background and no problems on this issue have been identified during operations to date (April 1985). The staff has independently confirmed the detectability of the red bulbs on the red background through interactions with the NRC resident inspector at Limerick. Based on this data, the staff considers this issue resolved.

18.1.10 Coordination of Control Room Improvements With Other Programs

The staff's SER on this issue requested that the licensee provide additional, detailed descriptions of these activities for evaluation. Some of these activities are described in the licensee's supplement to the Summary Report.

The licensee's supplement states that the objective of the Limerick Control Room Design Review Validation was to determine whether functions allocated to the control room operating crew could be effectively accomplished within the structure of the Transient Response Implementation Procedures (TRIP) and the improved design of the control room. Further, as part of the validation of the control room improvements, it was possible to determine if the improvements

created additional discrepancies, and to identify discrepancies not previously noted. The validation methodology involved three phases: preparation, walk-through/talk-through, and documentation of results.

The licensee's supplement describes the preparation phase as the development of the data collection guidelines and the analysis of the TRIP procedure flow diagrams. The analysis of the TRIP procedures was done to ensure that the validation effort examined the appropriate steps and contingencies of the procedures while minimizing the redundancies. This ensured that the main flow and branches to a TRIP were examined.

Also, the T-200 Series of Procedures were reviewed for control room actions, and those actions applicable to the control room were also walked through. The actual validation activity was conducted via walk-through and talk-throughs using the enhanced control room mockup.

Walk-throughs were performed by operators. The operators were briefed on the purpose of the validation and also briefed on the enhancements and changes to the control room configuration. Although the Safety Parameter Display System (SPDS) was not available, black and white hard copies of computer generated display formats were used as information sources during the walk-throughs. The SPDS is a subsystem within the Emergency Response Facility Data System (ERFDS). Also, communications among operators within the control room and from control room operators to floor operators were synthesized during the walk-throughs.

Data was collected on the walk-throughs by two human factors consultants and a human factors specialist. Each control room operator was accompanied by a data taker using an operator activity sheet to record movements from station to station, and a comment sheet to record comments by operators and to record observer notes.

Subsequent to the walk-throughs/talk-throughs, the data collected was reviewed, analyzed, and the results documented. A critique checklist based on statements drawn from NUREG-0700 (Section 6 Items) that applied to walk-through operations was used to evaluate the collected data. The potential discrepancies noted during walk-throughs/talk-throughs were reviewed, analyzed and where necessary, categorized as HEDs.

The licensee's supplement also states the results from the validation effort. Briefly, these were:

- no excessive duplication of instrumentation was observed,

- the improved design of the control room was validated as operators had no difficulty in locating controls and indicators. However, the need for additional modifications were also defined,

- the selection of prospective displays for ERFDS seemed to be well suited to the needs of operators,

- communications were found to be adequate for emergency conditions,

- the TRIP procedures were found to be effective.

Subsequently in a letter dated April 12, 1985, the licensee provided information on the coordination of the following NUREG-0737 initiatives:

1. Emergency Operating Procedures
2. Accident Monitoring Instrumentation - R.G. 1.97
3. Safety Parameter Display System (SPDS)
4. Emergency Response Facilities
5. Detailed Control Room Design Review

In addition, the licensee states:

> "During the final validation effort the coordination of the above system/facilities was completed. The actual validation was a walkthrough/talkthrough of the Emergency Operating Procedures using hardcopy displays for SPDS on a full-scale enhanced mockup of the Limerick Unit #1 control room. This walkthrough/talkthrough identified areas where a control room operator would require communications with the Emergency Response Facilities and the operator described what control room communications equipment would be utilized to complete this task. In addition HED SI1-06, (Reg. Guide 1.97 instrument enhancements) was implemented on the full scale mockup prior to this walkthrough/talkthrough. The operators could quickly identify these environmentally qualified instruments during these walkthrough/talkthroughs.
>
> This final validation was an effective method of coordinating the integration of the NUREG-0737 Supplement 1 items. Any discrepancies identified during these walkthrough/talkthroughs were classified as a HED and assessed in the Final Report Supplement No. 1."

In evaluating the HEDs identified by the licensee during the final validation, the staff noted that the HEDs did relate to the above defined initiatives. Based on the data provided by the licensee, it appears that the final validation did serve as a mechanism to coordinate the emergency response initiatives. However, as the staff has not completed its review of the System Function and Task Analysis portion of the DCRDR, the staff reserves its final evaluation of the coordination of control room improvements with other initiatives until this work has been evaluated by the staff.

18.1.11 Staff's Conclusions on DCRDR

As a result of its review the staff concludes that PECO's Limerick, Unit 1 DCRDR is meeting the NUREG-0737, Supplement 1 requirements for most of the work completed, but the staff review of the DCRDR is incomplete. The incomplete portions of the DCRDR consist of:

1. The use of function and task analysis to identify control room operator tasks and information and control requirements during emergency operation as submitted in the licensee's June 28, 1985 letter.

2. A comparison of display and control requirements with a control room inventory to identify missing displays as submitted in the licensee's June 28, 1985 letter.

3. A human-factors survey of the SPDS.

From our review of information provided by the licensee, the staff concludes that the following parts of condition 2.C.(8)(a) to License NPF-27 have been met and are no longer necessary:

1. Task Analysis and Control Room Inventory.

2. Control Room Survey.

3. Control Room Enhancements.

Also based on a review of the information provided by the licensee, the staff arrived at the following conclusions for specific elements of the DCRDR:

Control Room Survey

All of the HEDs defined by the NRC Audit Team during the in-progress audit have been incorporated into the review process, assessed by the licensee, and the results of the assessment are acceptable to the staff.

Assessment of HEDs

1. The correction of HED SI4-04 has been completed and implemented prior to operation at a power level greater than five percent of rated power.

2. A reassessment of HED SPV-07 is not needed as its correction has been completed.

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

The data provided by the licensee and independent confirmations by the staff verified the use of a solid red background as an enhancement to areas of the workbench and panels.

18.2 Safety Parameter Display System

All holders of operating licenses issued by the Nuclear Regulatory Commission (licensees) and applicants for an operating license (OL) must provide a Safety Parameter Display System (SPDS) in the control room of their plant. The Commission approved requirements for the SPDS are defined in Supplement 1 to NUREG-0737 (Ref. 7).

A. Background

By letter dated March 18, 1985 (Ref. 8), the licensee notified the NRC that General Electric's Emergency Response Information System (ERIS) SPDS displays cannot be released for use by operators because the Validation Testing process of the display's software is not complete. In response to a license condition that required an operational SPDS by April 1, 1985, the licensee proposed an interim SPDS. Upon evaluating the limited data on the interim SPDS, the staff concluded that it did not endorse the use of the interim SPDS. This conclusion, along with the reasons, was transmitted to the licensee by letter dated March 27, 1985 (Ref. 9). The licensee then proposed an amendment to License NPF-27 in a letter dated March 29, 1985 (Ref. 11).

B. Evaluation

The staff's safety evaluation of Limerick's SPDS is reported in SSER-3. This supplemental safety evaluation reports our review of the proposed amendment to the license and on the resolution of staff concerns expressed in SSER-3. The Regulatory requirements stated in NUREG-0737, Supplement 1 (Ref. 7) served as the basis for the staff's evaluation. In addition, we present our results in a format which conforms with SSER-3.

18.2.3 Parameter Selection

The staff's SSER-3 identified several concerns with the variables selected for display and associated with the Reactivity Control Safety Function. We made a conditional acceptance of these variables subject to a confirmatory review of information to be provided by General Electric. This information was received, reviewed and found acceptable. The results from the staff's review are presented in Reference 10.

In SSER-3, we requested the licensee to add to the SPDS a variable on containment radiation. The purpose of this variable is to identify the status of the Radioactivity Control Safety Function during periods when containment is isolated. In Reference 8, the licensee states that the Radioactivity Control Function of the SPDS is provided by the Radiation Meteorological Monitoring System (RMMS) which is operable and functionally available for use by operators. Further, the licensee states that containment radiation is monitored and displayed by RMMS to provide the status of the Radioactivity Control function during periods when containment is isolated. Based on this data, the staff's concerns on the Radioactivity Control Safety Function are resolved.

18.2.5 Human Factors Program

The staff requested additional data from General Electric on color coding of information in the display formats. The Limerick SPDS was also subject to the results from this review (SSER-3). This information was received, reviewed and found acceptable. The results from the staff's review are presented in Reference 10.

18.2.9 Implementation Plan

The staff conditioned Limerick's license to ensure an operational SPDS by April 1, 1985. By letter dated March 18, 1985, the licensee notified the NRC that the SPDS would not be operational by April 1 because testing of the system was incomplete. In this letter, the licensee proposed an interim SPDS which was evaluated by the staff and found to be unsatisfactory as an alternative.

In a letter dated March 29, 1985, the licensee requested an extension in the operability date to permit completion of the validation process for the SPDS. The letter states that final activities in the SPDS validation process must be performed during the Power Ascension Test Program at power levels up to and including 100 percent of rated power. This effort involves testing the parameter validation algorithms at defined plant power levels. Also, this effort includes refining system constants and performing reasonableness checks which

compare the SPDS display data with the hardwired plant instrumentation. Based on the staff's knowledge of the validation algorithms (Ref. 9), we concur that these activities are needed to ensure that reliable SPDS data are presented to operators.

The licensee's letter of March 29 also states that operators have been trained on the plant simulator to respond to transients and emergency conditions without the SPDS being available as required by NUREG-0737, Supplement 1. All operators licensed at Limerick have been thoroughly trained in the use of the BWR Owners' Group symptomatic emergency procedures. These procedures have been implemented through the development of the symptom oriented Limerick Transient Response Implementation Plan (TRIP's). In addition, all analog Control Room Reg. Guide 1.97 instrumentation has received special yellow highlighting to enable the operator to identify it for use in implementation of the TRIP's. The letter states that these instruments provide the information that is needed by operators to quickly and reliably assess the safety status of the plant.

Furthermore, the licensee proposes a schedule that the SPDS be initially operable within 30 days after the completion of the 100-Hour Warranty Run at 100 percent of rated power. Based on the need to validate algorithms during power ascension and on the fact that operators are trained to assess the safety status of the plant without an SPDS available, the staff finds that this schedule is acceptable. However, we request that the licensee provide the NRC with a summary of the problems encountered and the resolutions implemented in making the SPDS operable during the power ascension tests. This summary should be provided to the NRC upon completion of the 100-Hour Warranty Run at 100 percent of rated power. Also, we request the licensee to report results from the field verification tests of the SPDS. The purpose of these tests is to verify that the system is properly installed and will function and perform per requirements, in the plant environment.

18.2.10 Staff's Conclusions on SPDS

Based on the staff's review of the information provided by the licensee, the staff concludes:

1. The variables selected for the Reactivity Control Safety Function are acceptable.

2. Containment radiation is being monitored by the Radiation Meteorological Monitoring System. This system is operable and available for operators to evaluate the Radioactivity Control Safety Function. The staff's concerns on the Radioactivity Control Safety Function for the SPDS are resolved.

3. The color coding of information within display formats is acceptable.

4. The proposed revised schedule delaying operability of the SPDS until it has been fully validated is acceptable to the staff. However, the staff requests the licensee to provide a summary of the problems encountered and resolutions implemented in making the SPDS operable during the power ascension tests. Also, we request the licensee to report results from the field verification tests of the SPDS.

REFERENCES

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

2. Letter from J. S. Kemper, Philadelphia Electric Company, to A. Schwencer, NRC, Subject: "Limerick Generating Station, Units 1 and 2, Limerick Control Room Design Review, Final Report, Supplement 1," dated November 2, 1984

3. Letter from J. S. Kemper, Philadelphia Electric Company, to A. Schwencer, NRC, Subject: "Limerick Generating Station, Units 1 and 2, Satisfaction of License Condition 2.C.(8)(a), Detailed Control Room Design Review; Control Room Enhancements," dated March 21, 1985.

4. Letter from J. S. Kemper, Philadelphia Electric Company, to A. Schwencer, NRC, Subject: "Limerick Generating Station, Units 1 and 2, Detailed Control Room Design Review," dated April 4, 1985.

5. Letter from A. Schwencer, NRC, to E. G. Bauer, Jr., Philadelphia Electric Company, Subject: "Request for Additional Information - Limerick," dated March 14, 1985.

6. Letter from J. S. Kemper, Philadelphia Electric Company, to A. Schwencer, NRC, Subject: "Limerick Generating Station, Control Room Design Review," dated April 12, 1985.

7. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability," U.S. NRC Report NUREG-0737, Supplement No. 1, January 1983.

8. Letter from J. S. Kemper, Philadelphia Electric Company, to A. Schwencer, NRC, Subject: "Limerick Generating Station, Unit 1, Safety Parameter Display System, ERFDS, Reg. Guide 1.97 Displays," dated March 18, 1985.

9. Letter from A. Schwencer, NRC to E. G. Bauer, Jr., Philadelphia Electric Company, Subject: "Safety Parameter Display System for Limerick," dated March 27, 1985.

10. Letter from C. O. Thomas, NRC, to G. G. Sherwood, General Electric Company, Subject: "Draft SER On the Safety Parameter Display System for GESSAR II," dated December 18, 1984.

11. Letter from J. S. Kemper, Philadelphia Electric Company, to H. R. Denton, NRC, Subject: "Limerick Generating Station, Unit 1, Safety Parameter Display System," dated March 29, 1985.

12. Letters from J. S. Kemper, Philadelphia Electric Company, to W. R. Butler, NRC, Subject: "Limerick Control Room Design Review," dated May 24 and June 10, 1985.

13. Letter from J. S. Kemper, Philadelphia Electric Company, to W. R. Butler, NRC, Subject: "Limerick Control Room Design Review Final Report Supplement No. 2," dated June 28, 1985.

APPENDIX A

CHRONOLOGY

February 15, 1985	Letter from licensee on remote shutdown system.
March 6, 1985	NRC request for information on redundancy of the remote shutdown system.
March 18, 1985	Letter from licensee on safety parameter display system (SPDS).
March 27, 1985	NRC letter on SPDS verification.
March 21, 1985	Letter from licensee confirming that detailed control room design review (DCRDR) condition 2.C.(8)(a)(3) had been met.
March 25, 1985	Licensee letter providing information on the remote shutdown system.
March 29, 1985	Letter from licensee requesting amendment to condition 2.C.(8)(b) for SPDS operability.
April 4, 1985	Letter from licensee providing status of implementation of DCRDR condition 2.C.(8)(a).
April 9, 1985	Licensee letter providing information on significant hazards considerations associated with amended SPDS operability schedule.
April 10, 1985	Licensee letter on Unit 1 preservice inspection program relief requests.
April 12, 1985	Licensee letter providing information on condition 2.C.(8)(a) Part 2, DCRDR control room survey.
April 18, 1985	Licensee letter providing clarifications for remote shutdown system information.
May 1, 1985	NRC letter transmitting draft full power operating license.
May 10, 1985	Licensee letter on plans for completion of Unit 2.
May 17, 1985	Licensee letter commenting on draft full power operating license.
May 20, 1985	Licensee letter on containment inerting.

May 24, 1985	Licensee letter on control room design review.
May 29, 1985	Licensee letter responding to NRC letter of March 19, 1985 on Generic Letter 83-28 items.
June 7, 1985	Licensee letter describing status of Generic Letter 83-28 items.
June 10, 1985	Licensee letter on control room design review.
June 10, 1985	NRC letter transmitting revised draft full power operating license.

APPENDIX H

This Supplement No. 5 to the SER is a product of the NRC staff. The NRC staff members listed below were principal contributors to this report.

Name	Title	Branch
M. Hum	Materials Engineer	Materials Engineering
F. Eltawila	Senior Containment Systems Engineer	Containment Systems
R. Stevens	Reactor Engineer (INSTR)	Instrumentation and Control Systems
D. Kubicki	Fire Protection Engineer	Chemical Engineering
J. Sears	Emergency Preparedness Specialist	Emergency Preparedness
L. Crocker	Leader, Management Technology	Licensee Qualifications
L. Beltracchi	Senior Human Factors Engineer	Human Factors Engineering
O. Rothberg	Mechanical Engineer	Mechanical Engineering
C. Nichols	Senior Nuclear Engineer	Meteorology and Effluent Treatment

APPENDIX N
AMENDMENT NUMBER 1

SAFETY EVALUATION SUPPLEMENT ON
PRESERVICE INSPECTION RELIEF REQUEST

I. INTRODUCTION

This evaluation is in addition to the staff review reported in the corresponding Appendix in SSER 3. In a letter dated April 10, 1985, the licensee requested relief from other American Society of Mechanical Engineers (ASME) Code preservice examination requirements that it determined to be impractical to perform and provided a supporting technical justification pursuant to 10 CFR 50.55a(a)(3)*. The staff review of the information in this letter is discussed in the following paragraphs of this report.

II. TECHNICAL REVIEW CONSIDERATIONS

For an overall discussion of the bases of the staff review of the Limerick Unit 1 Preservice Inspection (PSI) Program and requests for relief from impractical requirements see Appendix N in SSER-3, Part II.

III. EVALUATION OF RELIEF REQUESTS

The licensee requested relief from specific Preservice Inspection (PSI) requirements in a submittal dated July 17, 1984. The licensee requested relief concerning additional subjects and revised or deleted other requests in submittals dated August 7, 1984, August 23, 1984, August 28, 1984, and August 30, 1984. These submittals contain a description of the issues, a detailed list of components for which relief was 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 was requested (Attachment 5 of the July 17, 1984 submittal), and a justification for the relief requests. The evaluation of these submittals is presented in SSER-3, Appendix N.

In a letter dated April 10, 1985, the licensee requested relief concerning additional subjects and revised or updated some of the requests evaluated in SSER 3. This letter also contains a description of the new issues, Revision 2 of the Component Summary Table, Revision 1 of the Safety Impact Summary, and justification for the relief requests. Many of the changes are editorial in nature and were intended to clarify previous submittals. Based on the information submitted by the licensee and the staff's review of the design, geometry, and materials of construction of the components, the staff has determined that certain preservice requirements of the ASME Boiler and Pressure Vessel Code, Section XI are impractical to perform. The licensee has demonstrated that either (i) the proposed alternatives would provide an acceptable level of quality and

*In the 1984 Edition of 10 CFR 50, this paragraph was designated as 50.55a(a)(2).

safety, or (ii) compliance with the specified requirements of this section would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3), conclusions that these preservice requirements are impractical are justified as follows.

Unless otherwise stated, reference to the Code refers 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. Since the relief requests discussed below are subsets of the categories of requests evaluated in SSER-3 only the supporting information (category title, Code requirement, etc.) related to these specific SSER-5 items is presented below.

A. <u>Category B-D, Reactor Vessel Nozzle Weld (Relief Request Number 5)</u>

<u>Code Requirements</u>: Examination Category B-D, Item Number B3.90. Table IWB-2500-1 in the Winter 1980 Addenda of Section XI requires a 100% volumetric examination of the nozzle to vessel welds in the reactor vessel.

<u>Code Relief Request</u>: Relief is requested from performing 100% of the Code-required volumetric examination. Additional information was provided for the nozzle to vessel welds in Relief Request 5.

<u>Reason for Request</u>:

The design of the reactor vessel includes nozzles that prevent automated ultrasonic examination of limited areas on some of the nozzle-to-vessel welds. In Attachments, 1, 2, and 3 of the April 10, 1985 letter, the licensee updated the description of the examination coverage, percent complete, and the obstructions for each nozzle-to-vessel weld addressed in Relief Request Number 5. The total number of nozzle-to-vessel welds has increased from 1 to 4 welds identified as N2B, N2G, N4B and N4D.

The licensee has provided sketches showing the design configuration and dimensions of the other three reactor vessel nozzles requiring ultrasonic examination under Examination Category B-D. The factors limiting the examination of these welds are the radius on the nozzle side of the weld centerline causing the transducer to lift off the surface and lose contact or interference due to the placement of another nozzle. The examination was performed from one side of the weld only based on ASME Section V, Article 4, paragraph T-441.4.4. The licensee estimated that between 69% and 75% of the four nozzles were fully examined.

<u>Staff Evaluation</u>: The staff has determined that the volumetric examination of the additional welds identified by the licensee is impractical to the extent required by the Code because of the design configuration of the nozzles. On the basis of the additional information contained in the licensee's letter related to the increase in the number of welds included in the relief request the staff reaffirms its SSER-3 conclusions for Relief Request No. 5 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 and 13)

Code Requirement:

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. Additional information was provided for the Class 1 and Class 2 pipe welds in Relief Request Numbers 6 and 13.

Reason for Requests: The design of Class 1 and Class 2 piping systems has welded joints, such as, pipe-to-fitting and pipe-to-component, which physically prevents all or part of the required Section XI examinations from the fitting or component side of the weld specified. The licensee has identified the piping system welds with obstructions, identified the obstruction, and estimated the percent of volume coverage.

In Attachments 1 and 2 of the April 10, 1985 letter, the licensee updated the description of the examination coverage, percent complete, and the obstructions for some piping welds addressed in Relief Request Numbers 6 and 13. The total number of ASME Code Class 1 welds (Relief Request Number 6) has increased from 59 to 67 welds. The total number of ASME Code Class 2 welds (Relief Request Number 13) has been increased from 53 to 64 welds.

Staff Evaluation: The staff has determined that the volumetric examination of the additional welds identified by the licensee is impractical to the extent required by the Code because of the design of the piping systems. On the basis of the information contained in the licensee's letter related to the increase in the total number of welds included in Relief Request Numbers 6 and 13 and the alternative examinations performed during construction the staff reaffirms its SSER-3 conclusions that the limited Section XI ultrasonic examinations, the augmented surface examinations, the volumetric examinations performed during fabrication, the hydrostatic test, and the licensee's Safety Impact Summary, as revised, 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 the support welds.

Reason for Request: In the initial submittal the licensee stated that there are three integrally welded support attachments per reactor recirculation pump for a total of six supports. In Attachments 1 and 2 of the April 10, 1985 letter, the licensee revised the description of the component and states that there are four integrally welded support attachments per reactor recirculation pump for a total of 8 supports included in this relief request.

Staff Evaluation: On the basis of the information contained in the licensee's letter that revises the description of the component and the increased number of support welds included in the relief request the staff reaffirms its SSER-3 conclusion that the fabrication examinations (radiography and surface examinations of the pump pressure boundary castings plus visual examinations of all fittings) and the licensee's Safety Impact Summary, as revised, 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 Request Number 10)

Code Requirement:

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 four-inch nominal pipe size.

Code Relief Request: Relief is requested from performing the Code-required visual examination of the internal surfaces. Additional information was provided for the valve bodies in Relief Request Number 10.

Reason for Request:

The total number of ASME Code Class 1 valve bodies (Relief Request Number 10) has decreased from 69 to 68 valves.

Staff Evaluation: The information contained in the licensee's letter reducing the number of valve bodies addressed by the request does not change the staff's conclusions reported in SSER 3 granting the relief requested by Request No. 10 for the remaining 68 valves.

E. Class 2, Category C-A Pressure Vessel Welds (Relief Request Number 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. The examinations shall cover at least 20% of each circumferential weld, uniformly distributed among three areas around the vessel circumference.

Code Relief Request: In the Preservice Inspection Program, the licensee committed to perform a volumetric examination of 100% of each vessel shell

and circumferential weld. Relief is requested from performing the volumetric examinations based on this commitment. Additional information was provided for the pressure vessel welds in Relief Request Number 21.

Reason for Request:

Joint configurations and other components prevented 100% examination coverage. The total number of vessel welds (Relief Request Number 21) in the "B" RHR heat exchanger has decreased from 3 to 2 welds. The licensee states that the ultrasonic examination coverage achieved during the preservice inspection exceeds the ASME Code requirements. Since the Code coverage requirement has been met, this relief request only involves the licensee's commitment to perform augmented examinations.

Staff Evaluation: The information contained in the licensee's letter reducing the number of welds addressed by the request does not change the staff's conclusions reported in SSER 3 granting the relief requested by Request Number 21. Granting of relief pursuant to 10 CFR 50.55(a) is not necessary because the licensee met the Code requirement during the preservice inspection and these requirements provide an acceptable level of quality and safety. Since the requirements of the regulation were completed for this examination category, the staff finds that the licensee does not have to complete the commitment to exceed the requirements of the Code.

F. Class 2, Category C-C, Integrally-Welded Support Attachments to Vessels and Category C-E-1, Integrally-Welded Support Attachments to Piping (Relief Request 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.

Examination Category C-E-1 of Table IWC-2520 in the Section XI Code requires surface examinations of piping integrally-welded attachment welds.

Code Relief Request: For Examination Category C-C, relief is requested from performing 100% of the Code-required surface examination on RHR heat exchanger support attachment welds. For Examination Category C-E-1, relief is requested from performing 100% of the Code-required surface examination on piping support attachment welds. Additional information was provided for Relief Request Numbers 12 and 16.

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.

The total number of integrally welded support attachments to vessels (Relief Request Number 12) has increased from 15 to 16 welds.

The total number of integrally welded support attachments to piping (Relief Request Number 16) has increased from 141 to 149 supports. During

the preservice inspection, some examinations were partially obstructed by additional hardware welded to the integral attachment or completely obstructed (e.g., eleven penetration supports encased in concrete). Construction nondestructive records for full penetration integral attachments were used, where possible, to supplement the preservice examination.

Staff Evaluation: Paragraph IWC-2100 of Section XI of the ASME Code permits the substitution of shop and field examinations for on-site preservice examinations provided that such examinations are conducted under conditions and with equipment and techniques equivalent to those expected to be used during subsequent inservice examinations and the shop and field examination records are in a form consistent with the preservice examination records.

On the basis of the information contained in the licensee's letter related to the increase in the number of integrally welded support attachments included in the relief requests the staff reaffirms its SSER-3 conclusions regarding Relief Requests Nos. 12 and 16 that the limited Section XI surface examinations supplemented by the fabrication examinations and the licensee's Safety Impact Summary, as revised, demonstrate an acceptable level of preservice structural integrity.

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. Additional information was provided concerning the number of pressure retaining welds.

Reason for Request:

The total number of ASME Code Class 2 pressure retaining welds in pumps has increased from 40, as evaluated in SSER-3, to 56 welds. Welds on the RHR and Core Spray Pumps received a limited preservice volumetric examination due to joint configurations (i.e., fitting-to-component) or to being encased in concrete. The licensee has identified the welds with limited examinations, identified the obstructions, and estimated the percent of volume coverage in Revision 2 of the Component Summary Table contained in the April 10, 1985 submittal. The licensee also states that the 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 pressure tests. Shell leakage can be detected at the foundation construction joints.

Staff Evaluation: On the basis of the information contained in the licensee's letter related to the increase in the number of welds included in the relief request the staff reaffirms its SSER-3 conclusions that the

limited Section XI examinations supplemented by the fabrication examinations and the licensee's Safety Impact summary, as revised, demonstrate an acceptable level of preservice structural integrity.

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

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

Code Relief Request:

Relief is requested from performing 100% of the required surface examinations of four bolting sets for valves as listed in the Component Summary Table, Revision 2 as included in the licensee's April 10, 1985 submittal.

Additional information was provided concerning the number of bolts in Relief Request Number 24.

Reason for Request: The total number of bolts in valves (Relief Request 24) has decreased from 9, as evaluated in SSER-3, to 4 bolting sets.

Staff Evaluation: The information contained in the licensee's letter related to the decrease in the number of bolting sets addressed by Relief Request 24 does not change the staff's conclusions reported in SSER 3 granting the relief requested for the remaining four bolting sets.

K. Class 1, Category B-J, and Class 2, Categories C-F and C-G, Pressure Retaining Welds in Piping (Relief Request Numbers 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. Additional information was provided concerning the number of pressure retaining welds in piping.

Reason for Request:

For geometric reflectors, the information not recorded on a consistent basis was the circumferential location 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.

The total number of ASME Code Class 1 piping welds (Relief Request Number 25) was increased from 37 to 42 welds.

The total number of ASME Code Class 2 piping welds (Relief Request Number 26) was increased from 70 to 86 welds.

Staff Evaluation: On the basis of the information in the licensee's letter related to the increase in the number of piping welds included in the relief requests the staff concludes that these relief requests are acceptable for PSI because the licensee 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 the inservice examinations.

L. Class 2, Category C-D, Pressure Retaining Bolting Exceeding 1-inch in Diameter (Relief Request Number 28)

Code Requirement: Examination Category C-D requires that a pressure vessel bolting exceeding 1 inch in diameter be visually and either surface or volumetrically examined. Bolting may be examined either in place under tension, when the connection is disassembled, or when the bolting is removed. Visual examinations shall cover 100% of the bolts, studs, nuts, bushings, and threads in base material and flange ligaments between threaded stud holes. Surface and volumetric examinations shall cover 10% of the bolting components, threads in the base material and flange ligaments between threaded stud holes for each bolted joint, but not less than two bolts or studs per joint and shall cover 100% of the bolting per inspection interval. The examinations shall include bolts, studs, nuts, bushings, washers and threads in the base material and flange ligaments between threaded stud holes. Bushings, threads and ligaments in the base material and flanges are required to be examined only when the connection is disassembled.

Code Relief Request: Relief is requested from performing the required visual examinations of 128 nominal diameter 1 1/8 inch bolts and a surface examination 10% of the bolting components in the two RHR Heat Exchangers.

Justification: The integrity of the RHR heat exchanger pressure retaining bolting has been verified by construction code testing requirements. Examinations were performed in accordance with the material specification and that edition of ASME Section III in effect at the time of procurement. Visual examinations were performed on the threads, shanks and heads (where applicable). Surface examinations were performed in either the finished bolting or the material stock just prior to threading. A hydrostatic pressure test was performed at 1.5 times the design pressure.

It is intended that the examinations performed by the manufacturer, serve as an acceptable alternative to the Section XI preservice requirements. The minimal safety impact of the incomplete Section XI preservice examinations described above is explained in the Safety Impact Summary. Sufficient system redundancy, leak detection capability, and alternative systems have been included in the plant design to assure plant safety.

All bolting included in this relief request is less than 2-inches in diameter. The first interval and successive inservice inspection programs will be based on the 1980 Edition (or later) including Addenda through Winter 1981 (or later) of the ASME Section XI Code. Therefore, only those pressure retaining Class 2 bolting exceeding 2-inches in diameter will be examined during each inservice inspection interval.

During the first interval, the RHR Heat Exchanger bolting will receive a visual examination (VT-1), either in place or disassembled, at a time of maintenance or concurrent with the scheduled volumetric examination of the shell welds.

Staff Evaluation: The regulation permits the licensee to meet the requirements of later editions of the Code referenced in 10 CFR 50.55a(b). The bolting included in this request is 1 1/8 inch nominal diameter. The 1980 Edition of Section XI of the ASME Code, which is referenced in 10 CFR 50.55a(b), only requires a visual examination of the entire bolted connection during pressure tests for ASME Class 2 bolting which is 2 inches and less in diameter. The objective of the visual examination is to locate evidence of leakage from pressure retaining components. The staff has determined that the licensee has performed examinations on the subject bolting during the hydrostatic tests that meet the requirements of approved editions of ASME Section XI. Therefore, the licensee has demonstrated an acceptable level of quality and safety by meeting the updated provision of the regulation.

The staff also reviewed the licensee's technical justification and determined that the licensee performed nondestructive examinations during construction that were equivalent or superior to the requirements of Section XI of the ASME Code committed to in the Preservice Inspection Program. Since ASME Section XI permits the substitution of shop and field examinations in lieu of on-site preservice examinations, the licensee has already performed examinations that exceed the requirements of approved editions of the Code.

IV. CONCLUSIONS

Based on the foregoing, pursuant to 10 CFR 50.55a(a)(3), the staff has determined that certain Section XI required preservice examinations are impractical. The licensee has demonstrated that either (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) 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 full power operation 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)(3), relief is allowed from these requirements which are impractical to implement.

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

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SEE INSTRUCTIONS ON THE REVERSE

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13. ABSTRACT (200 words or less)

In August 1983 the NRC issues its Safety Evaluation Report regarding the application for licenses to operate the Limerick Generating Station, Units 1 and 2 located on a site in Montgomery and Chester Counties, Pennsylvania.

Supplement 1 was issued in December 1983 and addressed several outstanding issues. SSER 1 also contains the comments made by the Advisory Committee on Reactor Safeguards in its interim report dated October 18, 1983. Supplement 2 was issued in October 1984. Supplement 3 was issued in October 1984 and addressed the remaining issues that required resolution before issuance of the operating license for Unit 1.

On October 26, 1984 a license (NPF-27) for Unit 1 was issued which was restricted to a five percent power level and contained conditions which required resolution prior to proceeding beyond the five percent power level. Supplement 4 issued in May 1985 addressed some of the technical issues and their associated license conditions, which required resolution prior to proceeding beyond the five percent power level. SSER 4 also contained the comments made by the Advisory Committee on Reactor Safeguards in its report dated November 6, 1984. This Supplement 5 to the SER addresses further issues. that require resolution prior to proceeding beyond the five percent power level.

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