NUREG-0991
Supplement No. 4

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

May 1985

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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 (NPF-27) 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. This Supplement 4 to the SER addresses some of those technical issues and their associated license conditions which require resolution prior to proceeding beyond the five percent power level. The remaining issues to be addressed prior to proceeding beyond the five percent power level will be addressed in a later supplement to this report. This Supplement 4 to the SER also contains 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.

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APPENDICES

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

In August 1983, the Nuclear Regulatory Commission staff (hereinafter referred to as the NRC staff) issued its Safety Evaluation Report (NUREG-0991) regarding the application by the Philadelphia Electric Company (hereinafter referred to as the 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 addressing and closing out numerous issues identified in the SER and in Supplement 1.

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.

The purpose of this Supplement No. 4 to NUREG-0991 (SER) is to update the SER by providing the evaluation and resolution of certain issues previously identified in the SER, its supplements and the license as requiring resolution prior to authorization to proceed beyond the five percent power level. Several other subjects are also addressed which were the subject of conditions in license No. NPF-27 or were discussed in the SER or its supplements.

The sections of this supplement are numbered the same as the corresponding section of the Safety Evaluation Report and Supplements No. 1 through 3. Each section is supplementary to and not in lieu of the discussion in the Safety Evaluation Report and Supplements No. 1 through 3 unless otherwise noted. Copies of this SER Supplement are available for inspection at the NRC Public Document Room, 1717 H Street NW, Washington, DC and at the Public Document Room at the Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464. They may be ordered from the sources indicated on the inside front cover of this report.

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

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

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

1.8 Outstanding Issues

The completion of the Independent Design Verification Program (IDVP) review and completion of the offsite emergency planning review, although not the subject of specific conditions in the license, were identified in SSER-3 as being relevant to the authorization of operations beyond the five percent power level. The IDVP review is discussed in Sections 17.5 and 3.9 of this report. The results of the offsite emergency planning review, as discussed in Section 13.3 of SSER-3, will be reported in a later supplement to NUREG-0991.

1.10 License Conditions

License No. NPF-27 contains 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 status of these license conditions is as follows. The conditions are numbered as they were in previous SER Supplements.

License Conditions		Section	Status*
(6)	Modifications to provide redundancy in remote shutdown system	7.4.2.3	R
(7)	Compliance with NUREG-0612 (Phase II, Heavy Loads Handling)	9.1.5	Closed
(13)	Personnel qualifications	13.1.2.2	Closed
(17)	Exception to the schedular requirements of the Standard Review Plan for certain fire protection items	9.5.1	Closed
(19)	Emergency Response capabilities		
	(a) Detailed control room design review	18.1	R
	(b) Safety Parameter Display System	18.2	R
	(c) Compliance with R.G. 1.97	7.5.2	Closed
(22)	Updating of environmental qualification records	3.11.3.3	Closed
(24)	Ultimate heat sink	9.2.5	Closed

*W - Awaiting information from licensee
R - Under NRC staff review

2 SITE CHARACTERISTICS

2.3 Meteorology

2.3.3 Onsite Meteorological Measurements

The staff discussed the onsite meteorological measurements program in Supplement 3 to the SER and stated as an outstanding issue that the adequacy of the meteorological program regarding data recovery would be confirmed after receipt and evaluation of at least one year of data and review of the applicant's response to the staff's improvement recommendation in the Emergency Response Appraisal regarding meteorological instrumentation inspection procedures and documentation of the results of each inspection.

In response, the licensee has submitted one year (October 15, 1983-October 14, 1984) of quarterly reports and data tapes to the staff containing meteorological information from instrumentation placed in service on October 15, 1983. With the exception of an extended period of down-time (April 4, 1984, - July 9, 1984) while the temperature systems were being calibrated, the joint data recovery of wind direction, wind speed and vertical temperature difference was about 97% from instrumentation on the primary meteorological tower (Tower No. 1). During the calibration period, the instrumentation on the backup meteorological tower (Tower No. 2) supplied joint meteorological data with a recovery greater than 90%. In addition, the licensee by letter dated March 6, 1985 has demonstrated that there is a high correlation between the data collected from Tower Nos. 1 and 2. Also the NRC staff reported in Inspection Report No. 50-352/84-61 that the licensee had developed an instrument inspection procedure (RT-11-00449) and had documented the results of each inspection. Therefore, the staff concludes that the licensee has demonstrated its capability to maintain adequate meteorological data recovery in accordance with Item III.A.2 of NUREG-0737, Supplement 1.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Rupture of Piping

As a part of the Independent Design Verification Program (IDVP) conducted by Torrey Pines Technology (TPT) for the Limerick Generating Station, Unit 1, Potential Finding Report (PFR) 019 was identified. As discussed in Section 17.5.4 of this report PFR-019 was resolved for the purposes of the IDVP and the disposition of specific jet loadings on ASME Class 1, 2, and 3 piping and supports is discussed in this section. The Limerick project pipe rupture analysis program required that jet impingement loads resulting from high energy line breaks be considered on all piping with diameters less than the ruptured pipe. The effects of jet impingement from smaller diameter pipes onto larger diameter pipes had not been considered. The applicant's basis for this exclusion is that pipe whip effects from smaller pipes onto larger pipes do not compromise the function of larger pipes and jet impingement loads can be shown to be less than pipe whip loads. The staff has reviewed this issue and has the following position:

Regulatory Guide 1.46, Footnote 14, NUREG 75/087 - SRP 3.6.1 -BTP APCSB 3-1 B.2.(2), and NUREG 0800-SRP 3.6.2 III.2 all clearly state that an unrestrained whipping pipe is considered capable of causing breaks in impacted pipes of smaller nominal pipe size and developing through-wall leakage cracks in impacted piping of equal or larger nominal pipe size with thinner wall thickness. Because of the differences in the nature of the loads from a whipping pipe and jet flow from a break or crack, the staff has not permitted the above guidelines for whipping pipes to be extended to jet impingement even though the equivalent static load from a jet is generally less than that from a whipping pipe.

In a letter from A. Schwencer to E. Bauer dated February 26, 1985, the staff requested the licensee to demonstrate that safe shutdown of the Limerick plant can be accomplished when the required jet impingement loads are included in the evaluation of target piping systems.

In response to the staff's request, the licensee submitted a letter dated March 19, 1985. The licensee has performed an extensive system interaction study to assess the potential jet impingement effects on the plant's ability to achieve a safe shutdown given the above staff position. A review of all previously postulated break locations identified in FSAR Section 3.6 was performed for both inside and outside containment. All potential target piping systems with diameter equal to or greater than the ruptured pipe were identified using piping layout drawings. A safe shutdown analysis was performed for each identified potential target. The licensee stated that of the 360 potential targets identified, only 24 targets were required to remain functional to assure safe shutdown. Of the 24 lines required for safe shutdown, there were 12 symmetrical cases. Out of

these 12 targets, 8 cases which would envelope the loads for all 24 targets, were analyzed to demonstrate that their function would not be impaired to the extent that safe shutdown of the plant could not be assured. Jet impingement loading on the target pipe was calculated using the methodology of BN-TOP-2, Rev. 2, which was previously referenced in the Limerick FSAR Section 3.6 for jet impingement analyses. Based on the results of its assessment, the licensee stated that the previously excluded target piping systems met the established project criteria for jet impingement loads on piping and pipe supports as described in the March 19, 1985 letter. Furthermore, the licensee has also performed a qualitative assessment of inclusion of jet impingement loads in the faulted loading combination and concluded that there was sufficient design margin to accommodate the combination of jet impingement load with SSE for the faulted condition for target piping and pipe supports. There were no hardware changes identified as a result of this jet impingement evaluation.

Based on a review of the information submitted by the licensee, the staff has determined that the licensee has provided adequate assurance that safe shutdown of the Limerick plant can be accomplished when the required jet impingement loads are included in the evaluation of target piping systems and, therefore, the staff considers this issue closed.

3.11 Environmental Qualification of Electrical Equipment Important to Safety and Safety Related Mechanical Equipment

3.11.3.3.1 Temperature, Pressure and Humidity Conditions Inside the Primary Containment

The NRC staff discussed the LOCA/MSLB temperature profile utilized for the Limerick equipment qualification program in Section 3.11.3.3.1 of Supplement 2 to the SER. THE NRC staff found the use of the generic NUREG-0588 profile for Limerick to be acceptable provided that the Equipment Qualification Review Records (EQRRs) were updated to reflect the use of the NUREG-0588 profile. Accordingly condition number 2.C(5) to license NPF-27 required that this be done prior to March 31, 1985.

The licensee has responded by letter dated March 28, 1985 addressing the EQRRs and indicating that they have been updated to reflect use of the NUREG-0588 profile. The NRC staff finds that the requirements of condition 2.C(5) in license NPF-27 have been met and the condition is no longer necessary.

7 INSTRUMENTATION AND CONTROLS

7.5 Safety-Related Display Instrumentation

7.5.2 Specific Findings (R.G. 1.97)

INTRODUCTION AND SUMMARY

As noted in Supplement No. 3 to the Limerick SER, the Philadelphia Electric Company (PECo) was requested by Generic Letter 82-33 to provide a report to the NRC describing how the Limerick post-accident monitoring instrumentation meets the guidance of Regulatory Guide 1.97 as applied to emergency response facilities. The licensee responded to the generic letter by letter dated April 15, 1983, referring to their Final Safety Analysis Report (FSAR). Additional information was provided by letter dated August 16, 1984.

The NRC staff's position as set forth in Supplement No. 3 to the SER was that pending the completion of the staff's review of the Limerick design for conformance to the guidance of Regulatory Guide 1.97, Revision 2, a condition to the license would require that modifications be completed to provide compliance with the Regulatory Guide unless the deviations identified by the licensee were reviewed and approved by the staff prior to startup following the first refueling outage. Accordingly, condition no. 2.C(8)(c) to license no. NPF-27 listed the following items as requiring justification: neutron flux, reactor water level, drywell sump level, drywell drain sump level, radiation level in circulating primary coolant, suppression spray flow and standby liquid control system tank level.

A detailed review and technical evaluation of the licensee's submittals was performed by EG&G Idaho, Inc., under contract to the NRC, with general supervision by the NRC staff. This work is reported by EG&G in their Technical Evaluation Report (TER), "Conformance to Regulatory Guide 1.97, Limerick Generating Station, Unit Nos. 1 and 2," dated October 1984 (Appendix P). The EG&G report addresses each of the items listed above as well as several others in the Limerick design. The NRC staff has reviewed this report and concurs with the conclusion that the licensee either conforms to, or is justified in deviating from, the guidance of Regulatory Guide 1.97 for each post-accident monitoring variable discussed in the report.

EVALUATION CRITERIA

Subsequent to the issuance of the generic letter, the NRC held regional meetings in February and March 1983, to answer licensee and applicant questions and concerns regarding the NRC policy on this matter. At these meetings, it was noted that the NRC review would only address exceptions and deviations taken to the guidance of Regulatory Guide 1.97. Further, where licensees or applicants explicitly state that instrument systems conform to the provisions of the guide, it was noted that no further staff review would be necessary. Therefore, the review performed and reported by EG&G only addresses exceptions and deviations to the guidance of Regulatory Guide 1.97. This Safety Evaluation addresses the

licensee's submittals based on the review policy described in the NRC regional meetings and the conclusions of the review as reported by EG&G.

EVALUATION

The NRC staff has reviewed the evaluation performed by EG&G as contained in the Technical Evaluation Report in Appendix P, and concurs with its bases and findings. The licensee either conforms to, or has provided an acceptable justification for deviations from, the guidance of Regulatory Guide 1.97 for each post-accident monitoring variable.

CONCLUSION

The NRC staff concludes, based on the NRC staff's review of the Technical Evaluation Report (Appendix P) and the licensee's submittals, that the Limerick Generating Station, Unit Nos. 1 and 2, design is acceptable with respect to conformance to the guidelines of Regulatory Guide 1.97, Rev. 2. The NRC staff, therefore, finds that the requirements of condition no. 2.C(8)(c) in license no. NPF-27 have been met and the condition is no longer necessary.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.5 Overhead Heavy Loads Handling System

The NRC staff's Safety Evaluation Report (SER) issued in August 1983 stated that a condition would be placed in the license requiring that before startup after the second refueling outage, the licensee would have made commitments acceptable to the NRC regarding the guidelines of Sections 5.1.2 through 5.1.6 of NUREG-0612 (Phase II, 9-month responses to the NRC generic letter dated December 22, 1980).

On February 6, 1984, the NRC staff transmitted by letter to the licensee a draft technical evaluation report (TER) of an NRC contractor's (EG&G Idaho, Inc.) review of the licensee's response to Phase II of the heavy loads program for Limerick. The NRC staff's letter requested a response to several identified concerns in the TER. By letter dated August 13, 1984, the licensee submitted Revision 3 to the Limerick Overhead Handling Systems Final Report including responses to the concerns identified in the draft TER by EG&G.

On October 26, 1984, license no. NPF-27 was issued with condition no. 2.C(19) requiring (a) commitments acceptable to the NRC staff regarding the guidelines of Sections 5.1.2 through 5.1.6 of NUREG-0612 (Phase II) and (b) the implementation of modifications and procedures required to fully satisfy the guidelines of Sections 5.1.2 through 5.1.6 of NUREG-0612 (Phase II). The licensee's submittal of August 13, 1984, addressed compliance with Sections 5.1.2 through 5.1.6 of NUREG-0612 and made commitments regarding the implementation of procedures for the control of heavy loads.

The staff has subsequently eliminated the need for further effort regarding compliance with the criteria of Phase II of NUREG-0612 on the basis of Phase I (Section 5.1.1) compliance and Phase II reviews to date. These Phase II reviews consisted of an evaluation of the responses for 12 randomly selected operating plants which formed a pilot program. The staff determined from these reviews that most of the risk associated with heavy loads handling has been resolved by implementation of Phase I. In addition, no further heavy loads handling concerns were identified from the pilot program reviews. It is, therefore, concluded that the objective identified in NUREG-0612 for providing "maximum practical defense in depth" is satisfied without the need for further action regarding Phase II and therefore, the condition 2.C(19) in license NPF-27 is no longer necessary.

9.2 Water Systems

9.2.5 Ultimate Heat Sink

Supplement No. 3 to the SER provided an evaluation of the ultimate heat sink which concluded that the Limerick Generating Station met the applicable General Design Criteria 2 and 4 with the exception of the period of operation up to five percent power.

Supplement No. 3 further stated that a plant procedure (which would be approved and implemented prior to exceeding 5% power) would provide guidance for the repair of the ultimate heat sink after damage by wind generated missiles. Condition 2.C(18) of the Limerick license NFP-27 requires this procedure to address "(a) the methods and resources for repair of spray pond piping, (b) operation of the spray pond in either of the closed cycle or once through cooling modes, (c) restoration of offsite power to the Schuylkill River makeup pumphouse and (d) the verification of the availability of portable pumps to pump water from the Schuylkill River to the spray pond pumphouse wetwells." On December 20, 1984, members of the staff visited the Limerick Generating Station in order to review these new procedures. The procedures are "symptom-based" and include special events. A new special event procedure (SE-9, "High Wind and Tornadoes") had been developed to be the base document for assuring cooling water in the event of damage to the ultimate heat sink. This procedure is entered only when high winds generate substantial quantities of airborne debris. However, the operators will receive advance warning of potential high winds from the load dispatcher in Philadelphia, Pennsylvania, where regional weather is continuously monitored. This advance notice is provided to allow time to prepare the electrical system for emergency demands for power.

Procedure SE-9 references procedure CPL-11 which gives guidance for the repair of damaged spray networks. Priority is placed on returning one of the spray networks, or equivalent, to service for one unit operation (two networks for two unit operation) by using available pipe and Dresser couplings and by scavenging parts from other networks. Procedure SE-9 is activated whether the damage is minor or constitutes a total failure of all networks. The staff concludes that this satisfies part (a) of condition 2.C(18).

Procedure SE-9 (in its Appendix B) provides instruction for the operator to prepare for once-through cooling operation, based on the amount of damage sustained by the spray networks, and also references Procedure S12.7A regarding use of the winter bypass return line to the spray pond. The winter bypass provides for circulation of water within the pool and bypasses the spray networks. Procedure SE-9 references other procedures which are concerned with other activities necessary for the once-through cooling mode of operation, such as the clearing of debris generated by failure of the cooling towers away from the makeup lines at the cooling towers and construction of an earthen dike in the event of the failure of the cooling tower basin retaining wall. The staff concludes that this satisfies part (b) of condition 2.C(18).

Power for the pumps at the Schuylkill River pumphouse is provided from the 2300 volt plant services switchgear. This switchgear can be fed using offsite power from either of the two plant substations via underground lines. The two substations are on approximately opposite sides of the site and therefore it seems unlikely that a tornado which could disable the spray pond networks and the cooling towers could also disable both substations. However, the procedures for once-through cooling also address the potential need to provide emergency power to the Schuylkill River pumphouse for the existing pumps or portable pumps (SE-9, Appendix C). The staff concludes that this satisfies part (c) of condition 2.C(18).

Procedures have been provided for the procurement of temporary water pumps, fittings, and airborne transportation, as necessary (SE-9, Appendix A). This

procedure includes a list of suppliers, the components which may be required from each supplier, and the method of obtaining these components. The licensee committed in the FSAR to annually verify these procedures which include contacting each supplier to verify the availability of the components or acceptable alternate components as necessary. The staff concludes that this satisfies part (d) of condition 2.C(18).

Based on the above, the NRC staff concludes that the licensee has met the requirements of condition 2.C(18) in License No. NPF-27 and has satisfactorily demonstrated compliance with General Design Criteria 2 and 4, with respect to providing compensatory measures to assure adequate plant cooling in the event of damage to the ultimate heat sink due to natural phenomena and missiles. Furthermore, the staff concludes that these procedures are acceptable and there will be no undue risk to the health and safety of the public. Based on the NRC staff's conclusion that the subject procedures are acceptable in meeting the requirements of the four parts of condition 2.C(18) and that there will be no undue risk to the health and safety of the public, the staff finds that such a license condition is no longer necessary.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

As discussed in Supplement 2 to the SER and in Item 1 of Attachment 2 to License NPF-27 the licensee was required to complete the following modifications prior to proceeding beyond the five percent power level.

(a) Install automatic sprinkler systems in Fire Area 41 (RECW Equipment Area) and 42 (Safeguard System Access Area).

(b) Provide additional automatic sprinkler system coverage in the NE corner of the Reactor Building, elev 283' (Fire Area 47A).

(c) Complete necessary modifications to the control structure fire protection system, elevations 332' and 350', to ensure that the standpipe hose stations are capable of flowing 100 gpm at 65 psi.

The licensee has responded by letter dated April 8, 1985 indicating that these specific actions have been completed. The NRC staff finds that the requirements of these parts of condition 2.C(6)(d) Attachment 2, Item 1 in license NPF-27 have been met and these parts of condition 2.C(6)(d) are no longer necessary.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Operation

13.1.2.2 Personnel Qualifications

The staff noted in Supplement 3 to the SER that the licensee had committed to furnish the results of oral and written examinations administered to the two prospective shift advisors and to train the operating shift crews in the role of the shift advisors.

By letter dated December 10, 1984, the licensee certified to the staff that both shift advisor candidates passed the oral and written examinations administered on September 21, 1984. A copy of the oral examination format and the written examination questions was enclosed with the letter. The staff has reviewed the material submitted by the licensee and finds the scope of the examinations acceptable. The licensee has satisfied the shift advisor certification requirements of the Limerick Unit 1 low power license.

The staff has also confirmed that the licensee has completed training the operating shift crews in the role of the shift advisors.

This item is closed and accordingly these portions of Attachment 3 to License NPF-27 addressing this issue are no longer necessary.

17 QUALITY ASSURANCE

17.5 Independent Design Verification Program

17.5.1 Background

The staff's interim report on the independent design verification program (IDVP) was included in Supplement 3 to the SER. This report presents the staff's final report as a result of the review of the IDVP.

Torrey Pines Technology (TPT), a Division of GA Technologies, Inc. was engaged by the Philadelphia Electric Company (PECo) to conduct an independent design review of Limerick Generating Station Unit No. 1 Core Spray System (CSS). PECo undertook this action to provide greater assurance of the adequacy of the Limerick design in view of design problems identified at other nuclear facilities. The core spray system was selected as a portion of the Limerick design to be reviewed which represented a sample of the design process employed by both the architect-engineer (Bechtel Power Corporation (BPC)) and the nuclear-steam-system-supplier (General Electric Company (GE)).

The TPT program plan was presented to the NRC at a May 9, 1984 public meeting in the NRC's Bethesda, Maryland office. In a letter to PECo dated May 15, 1984, the NRC found the program plan acceptable subject to minor programmatic clarifications and protocol modifications. The original program plan was changed to reflect the NRC clarifications and issued as Revision A, dated June 6, 1984. TPT initiated the independent design review on May 21, 1984. The NRC visited TPT's offices on July 24th and 25th and September 25th and 26th to monitor implementation of the program plan. TPT's final report of the review, dated November 1984, was received by the NRC in mid-December 1984. TPT presented a summary of the report at a meeting with the NRC and PECo on January 10, 1985. On January 15, 1985, the NRC visited BPC's offices to verify the results of corrective action completed following submission of the TPT report.

The NRC Staff's assessment of the TPT report and the subsequent corrective action is provided below.

17.5.2 TPT Evaluation Process

The TPT program consisted of six tasks, three tasks which evaluated the CSS design process and design adequacy, a fourth task for installation adequacy, a fifth task for processing of potential findings, and a sixth task for program management, administration and reporting, as follows:

- Task A: Design Procedure Review
- Task B: Design Procedure Implementation Review
- Task C: Technical Review
- Task D: Physical Verification Walkdown
- Task E: Processing of Potential Findings
- Task F: Administrative and Reporting

Task A evaluated compliance of the design procedures and controls with the NRC-approved Quality Assurance section of the FSAR and the requirements of 10CFR50, Appendix B. Task B evaluated CSS design documents for compliance with the established design procedures and controls identified in Task A. Task C then evaluated the technical adequacy of selected portions of the CSS with the NRC-approved criteria of the FSAR, while Task D evaluated the installation of selected portions of the system for conformance with design documents and specifications.

Tasks E and F dealt with administrative matters such as processing of potential findings and report writing. These tasks are not addressed in the TPT report and accordingly are not addressed in this SER.

When a deviation was identified, a potential finding report (PFR) was prepared. Each PFR was reviewed and evaluated through several steps before being ultimately classified as a finding, an observation, or as invalid. Essentially, a PFR was classified as a finding if a substantial safety hazard existed or if a generic deviation was indicated that might cause a substantial safety hazard. Specifically, the five criteria used by TPT for classifying PFRs as findings are summarized below:

1. Design margins had been reduced to the extent that allowable design limits were exceeded or that safety-related design requirements were not met.

2. An isolated procedural violation was of a nature that a substantial safety hazard could be created.

3. Repetitive procedural violations implied that a substantial safety hazard could be created.

4. Numerous similar deviations might exist such that a substantial safety hazard could be created (e.g., errors in commonly used calculational techniques).

5. Repetitive discrepancies resulting from unsuitable methods or errors, which in themselves were not a substantial safety hazard, suggest that other errors might exist which could create a substantial safety hazard.

Potential findings that were valid but that did not satisfy any of the above criteria were classified as observations. PFRs were classified as invalid, if, as a result of the PFR, additional information was provided to TPT to eliminate the concern. All PFRs, regardless of classification (finding, observation, or invalid), are contained in the TPT report. Where a PFR was classified as a finding, a corrective action report (CAP) was prepared by PECo. The CAPs are also included in the TPT report.

17.5.3 TPT Conclusions

TPT drew several conclusions from the independent review. With regard to adequacy of the design process, TPT stated:

"The design control procedural system and its implementation were effective in generating an adequate design, as demonstrated by the detailed technical review of selected portions of the CSS. The design generally complied with the NRC approved design bases and methodologies given in the FSAR. The deviations which were detected are expected to be accommodated within the margin of conservatism generally found throughout the design approach. Thus the final conclusions are based on the expectation that the CAPs will not result in hardware changes".

TPT summarized its conclusions with the following statement:

"In summary, based on the review which was performed of the design related procedures and their implementation in Task A and B, the technical review performed in Task C, the general trend of an adequate design process which appeared from these reviews, and the expectation that implementation of stated CAPs will not result in changes to the as-built design, it is judged that the design of the Limerick Unit 1 CSS is adequate."

In addition, at the January 10, 1985 meeting in Bethesda, the NRC questioned TPT as to whether the specific conclusion of design adequacy of the core spray system could be extended to include the overall Limerick 1 design process. TPT stated that, as a result of their review, they could state that the design process at Limerick 1 was adequate.

17.5.4 Assessment by NRC Staff

The NRC Staff performed an assessment of the TPT report and its findings to determine the significance of the findings and to assess the licensee's corrective actions. As part of the assessment, the Staff met with representatives of PECo, TPT, GE, and BPC on January 10, 1985 to discuss the TPT report and the NRC's assessment of TPT's findings and conclusions. As a result of this meeting, the NRC visited BPC's office in San Francisco on January 15th to review specific corrective actions. Details of the NRC's assessment of specific findings, observations, and corrective action plans are presented below:

Task A (Design Procedure Review)

There were no valid findings or observations resulting from this task.

Task B (Design Procedure Implementation Review)

The Task B review resulted in nine valid PFRs. Two of the nine were classified as findings, and seven as observations. In addition to these PFRs, one PFR was initiated in this Task which was eventually determined to be invalid.

PFR 009, classified as a finding, addressed the fact that audit checklists for one audit report contained some responses to questions which were not complete and some checklist questions which were not addressed. PECo's corrective action identified the extent of the problem, corrected identified deficiencies, and implemented a procedural change to prevent recurrence of the deficiency. The Staff finds the licensee's CAP on this matter to be acceptable.

PFR 026, classified as a finding by TPT, addressed the fact that 10 items were identified for which documents required by the GE design control program to demonstrate a design adequacy review were not available. GE did not agree that this was a finding, pointing out that most of these documents were created prior to June 1972, when retention was not required. Nevertheless, GE conducted a review to verify adequacy of the documentation. According to information provided by PECo and GE subsequent to receipt of the TPT report (at the January 10th meeting), this review has been successfully completed and the adequacy of documentation had been verified, with only minor editorial changes needed. The Staff has reviewed the nature of the changes performed by GE. Since the GE review did not impact either as-built hardware or design documents, the Staff finds the licensee's action to be acceptable.

The seven PFRs classified as observations and the one invalid PFR were reviewed by the Staff and the Staff concurs with their classification. Therefore further assessment by the Staff is not warranted.

Task C (Technical Review)

Task C resulted in the initiation of 22 PFRs, 6 of which were subsequently classified as invalid. Of the remaining 16 valid PFRs, 6 were classified as findings and 10 as observations.

PFR 014, classified as a finding, documented incomplete analytical qualification of reactor vessel core spray nozzles 5A and 5B. The initial vessel stress calculation performed by Chicago Bridge and Iron (CBI) on behalf of General Electric had not addressed the thermal stress ratchet requirement specified in the ASME Code. The code requirement had not been addressed because a change in the ASME Code, subsequent to issuance of the original specification for the vessel, had not been reflected back into the specification. PECo's corrective action addressed both the specific and generic aspects of the PFR. In particular, a 100% review of the vessel stress report and safe-end modification stress report was performed and the thermal ratcheting calculations were performed as applicable. All calculated stresses were within allowable levels for the core spray nozzels and other vessel nozzles for the Limerick vessel. The Staff finds the corrective action taken in this matter to be proper and sufficient.

PFR 016, classified as a finding, indicated that the pipe sleeve portion of the containment penetration X-16A had not been analyzed per ASME, MC Code requirements. Specifically, the core spray penetration did not satisfy ASME Code rules applicable to nozzles for primary upset loads (i.e., for the normal plus upset condition) nor were they enveloped by the stresses in the main steam and feedwater penetrations. PECo and BPC took issue with this finding, stating that a penetration was different than a nozzle and the general rules for containment vessels should apply, rather than the specific rules for vessel nozzles. Nevertheless, PECo undertook corrective action to deal with the specific analysis for penetration X-16A and also to confirm the adequacy of the engineering judgement that the main steam and feedwater penetration sleeve analyses are bounding for all piping systems with flued head attachments. At the January 10, 1985 meeting in Bethesda, PECo presented the results of these analyses. BPC indicated that the core spray analysis had been performed using the lower allowable stress levels indicated by TPT and nominal pipe wall thickness rather than minimum (ultrasonic testing (UT) had been performed to confirm the existence of

nominal thickness). BPC stated that: (1) the lower allowable levels were met in the core spray penetration, (2) the main feedwater analysis had also been revised, and (3) all containment penetrations with flued-heads were now verified as meeting the more conservative nozzle stress intensity limits. In reviewing the other flued-head penetrations, BPC used the lower allowable levels and minimum wall thickness, where possible. Where nominal wall thickness was needed to meet allowable limits, ultrasonic testing was performed to verify the as-built thickness. Since all flued-head penetrations were checked, the question of whether the main steam and feedwater penetrations are bounding is no longer relevant. During the subsequent visit to BPC on January 15, 1985, the Staff reviewed the analyses associated with these penetrations and found them acceptable.

PFR 017, classified as a finding, documents the lack of a thermal transient calculation for approximately six Class 1 valves supplied by General Electric that were installed in the core spray and residual heat removal systems. Subsequent analyses of these valves showed that significant margin remained after thermal fatigue was considered. The corrective action program addressed both specific and generic aspects of PFR 017 and is considered to be acceptable to the staff.

PFR 021, classified as a finding, concerned the ability of a one-inch instrument line to perform its safety function after it was subjected to jet impingement loads from a postulated core spray line break. The initial qualification of the line was based on a generic qualification using results of a test conducted on a similar line, however, TPT considered the generic qualification to contain errors and an unconservative extrapolation of test data. A subsequent analysis by BPC qualified the one-inch line. PECo addressed the generic aspects of this PFR along with the corrective action for PFRs 023 and 024, discussed below. The staff finds the corrective action taken in the case of PFR 021 to be acceptable.

PFRs 023 and 024, both classified as findings, identified errors and inconsistencies in the analysis that was used to demonstrate safe shutdown capability following postulated breaks in core spray lines. BPC agreed that there were specific areas in the analysis needing clarification or correction but did not agree that plant safe shutdown capability had not been demonstrated. Nevertheless, PECo proposed to take action to review and revise, as necessary, all safety evaluation calculations associated with jet impingement and to provide a description of the methodology of the analysis, including a discussion of how worst-case single failures are identified. At the meeting in Bethesda on January 10, 1985, PECo stated that the corrective action associated with this item had been completed and that no hardware changes were required but that minor changes in documentation had been incorporated. At the subsequent visit to BPC's offices on January 15, 1985, the Staff reviewed BPC's calculations. As a result of its review, the Staff concludes that the corrective action in this area is acceptable.

PFR 019, classified by TPT as invalid, and PFR 022, classified by TPT as an observation, involved design commitments with respect to design loading combinations for ASME 1, 2, and 3 piping or pipe support systems. PFR 019 involved jet impingement loadings on core spray pipe, while PFR 022 involved jet impingement loads associated with pipe supports. These PFRs were discussed with the licensee during the January 10, 1985 meeting in Bethesda and again during the Staff's

visit to BPC on January 15, 1985. The Staff's review concluded that jet impingement loads from all potential break locations had not been incorporated in the load combination for the faulted condition of the ASME Code stress analysis. BPC had extended the exemption for pipe whip considerations of SRP 3.6.2 (i.e., considering the sizes of the whipping and target pipe) to apply to jet impingement loading. BPC believed that this was a valid extension of the SRP exemption but the Staff disagreed and indicated that the jet loadings should have been considered. Since there was no breakdown in the functioning of BPC's design process in this area (BPC carried out the design in accordance with their intent and established practice), this issue is considered closed for purposes of the independent design review. Disposition of specific jet loadings on ASME Class 1, 2, and 3 piping and supports is discussed in Section 3.6.2 of this report.

PFR 032, classified as an observation, involved a BPC calculation which did not consider loads resulting from a postulated pipe break in the core spray line on the core spray reactor vessel nozzle. At the January 10, 1985 meeting the Staff requested TPT to elaborate on its conclusion that it was unlikely that this omission was a generic problem. TPT answered that the analysis had been performed by TPT which indicated the nozzle had considerable margin to withstand pipe whip loading. Further, TPT had reviewed the general configuration of other large lines and discovered they had pipe restraints even closer to the vessel than the core spray line, indicating even lower piping reaction loading. This reply satisfied the Staff's concern and no further action is required on this PFR.

The remaining observations and invalid PFRs associated with Task C were also reviewed by the Staff. The Staff concurs with the classification of these PFRs as observations or invalid and therefore, no further assessment by the Staff is warranted.

Task D (Physical Verification Walkdown)

There were three observations and one invalid PFR initiated in this task. The Staff concurs with the classification of these PFRs and therefore further assessment is not warranted.

17.5.5 Conclusion

On the basis of the Staff's review described above, the Staff concludes that the TPT design verification program provides an additional measure of assurance that the design process used in constructing Limerick Unit 1 complied with NRC regulations and licensing commitments. The specific matter of jet impingement loading on ASME Class 1, 2, and 3 piping of PFRs 19 and 22, is not the result of a breakdown in BPC's design process and is discussed further in Section 3.6.2 of this report.

19 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

During 1983, the Advisory Committee on Reactor Safeguards (ACRS) performed a review of the application for an operating license for the Limerick Generating Station Units 1 and 2. The results of that review were included in the ACRS interim report to the NRC Chairman dated October 18, 1983. The interim report and a discussion of it by the NRC staff are included in Supplement No. 1 to the Limerick Safety Evaluation Report. The interim report indicated that the ACRS believed that fuel loading and operation up to five percent power could be carried out without undue risk to the health and safety of the public.

During 1984, the ACRS continued its review of the Limerick application. The ACRS held a combined meeting of its Subcommittee on the Limerick plant and its Subcommittee on Reliability and Probabilistic Assessment on October 9-10, 1984, in Washington, D.C. A further meeting of the Subcommittee on Reliability and Probabilistic Assessment was held on October 20, 1984, in Los Angeles, California.

On November 1-3, 1984, the full Committee considered the Limerick application at its 295th meeting in Washington, D.C. A copy of the Committee's report, dated November 6, 1984, to the NRC Chairman is included in Appendix J to this Supplement to the SER. Because of the uncertain schedule for Unit 2, the Committee did not believe it appropriate to report on Unit 2 at this time. The Committee's report indicated that the Committee believed that, subject to the resolution of certain identified issues, there is reasonable assurance that the Limerick Generating Station Unit 1 can be operated at power levels up to 3293 MWt (100%) without undue risk to the health and safety of the public.

The Committee's November 6, 1984 report stated that the NRC and the industry should continue to work to develop methods which can be used to quantify seismic risk and to identify any seismic outliers which might exist. The Executive Director for Operations has established an NRC Working Group on Seismic Design Margins. This group has set up a panel of expert consultants who are developing methods and procedures to estimate the capacity of nuclear power plants to withstand earthquake ground motion beyond the design basis. The Working Group and its consultants have met and will continue to meet with the Committee to keep the Committee informed.

The Committee, in its November 6, 1984 report also recommended that Limerick receive special attention in the NRC staff's resolution of Unresolved Safety Issue A-17 (USI A-17), Systems Interactions in Nuclear Power Plants. The NRC staff has reviewed the Philadelphia Electric Company's efforts to identify systems interaction problem areas and has concluded that additional interaction studies at Limerick would not be likely to yield improvements that would result in a significant improvement in risk. The staff has considered all systems interactions studies performed to date, including the work for Limerick, as part of developing the resolution of USI A-17. Should the generic resolution indicate the need for plant-specific actions, the NRC staff plans to provide specific criteria and guidance as part of the resolution.

APPENDIX A

CHRONOLOGY
LIMERICK GENERATING STATION, UNITS 1 AND 2

October 26, 1984	Letter to licensee transmitting License No. NPF-27 for Limerick Unit 1.
November 2, 1984	Letter from licensee transmitting Supplement 1 to Control Room Design Review Final Report.
November 9, 1984	Letter from licensee responding to Generic Letter 84-11 on piping inspections for IGSCC.
November 14, 1984	Letter from licensee providing meteorological data for July-October 1984 period.
November 30, 1984	Letter from licensee providing primary reactor containment integrated leakage rate test final report.
December 10, 1984	Letter from licensee updating status of shift experience.
December 12, 1984	Letter from Torrey Pines Technology to licensee transmitting Independent Design Verification Program Final Report.
December 13, 1984	Letter from licensee transmitting Revision 5 of the Pump and Valve Testing Program Plan.
December 17, 1984	Letter from licensee responding to Generic Letter 84-23 on level instrumentation.
December 28, 1984	Letter from licensee on IDVP action items.
January 25, 1985	Letter from licensee on HPCI/RCIC turbine taper pin installation.
January 28, 1985	Letter from licensee on organizational changes in Electric Production Department for the station's operation phase.
February 13, 1985	Letter from licensee forwarding results of surveillance tests for TMI Item III.D.1.1 for primary coolant outside containment.
February 26, 1985	NRC request for information on jet impingement loads.
February 28, 1985	Letter from licensee forwarding semiannual effluent releases report.

March 6, 1985	Letter from licensee forwarding comparison of meteorological tower 1 with tower 2 data.
March 7, 1985	Letter from licensee forwarding revision 2 to solid radwaste process control program.
March 8, 1985	Summary of meeting on jet impingement loads.
March 15, 1985	Letter from licensee responding to Generic Letter 84-24 on environmental qualification of electrical equipment.
March 19, 1985	Letter from licensee on jet impingement loads.
March 19, 1985	NRC request for information on Generic Letter 83-28 (Salem ATWS) issues.
March 25, 1985	Letter from licensee on test operation of the main turbine.
March 28, 1985	Letter from licensee confirming that equipment qualification records have been updated to reflect use of NUREG-0588 temperature profile.
April 1, 1985	Letter from licensee on Salem ATWS responses (GL 83-28).
April 2, 1985	Letter from licensee forwarding revisions to safeguards contingency plan.
April 8, 1985	Letter from licensee confirming that requirements of condition 2.C(18) on the ultimate heat sink have been met.
April 8, 1985	Letter from licensee confirming that requirements of condition 2.c(6)(d) on fire protection have been met.
April 11, 1985	Letter from licensee on containment leakage testing requirement.
April 19, 1985	Letter from licensee forwarding revisions to security personnel training and qualification plan.

APPENDIX H

NRC STAFF CONTRIBUTORS AND CONSULTANTS

The Supplement No. 4 to the Safety Evaluation Report is a product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report.

Name	Title	Branch
E. Markee	Senior Meteorologist	Meteorology and Effluent Treatment
J. Milhoan	Chief, Licensing Section	Quality Assurance
G. Imbro	Senior Inspection Specialist	Quality Assurance
R. Parkhill	Inspection Specialist	Quality Assurance
J. Ridgely	Mechanical Engineer	Auxiliary Systems
L. Crocker	Leader, Management Technology Section	Licensee Qualifications
D. Kubicki	Fire Protection Engineer	Chemical Engineering
J. Joyce	Senior Reactor Engineer (INSTR)	Instrumentation and Control Systems
A. Masciantonio	Equipment Qualifications Engineer	Equipment Qualifications
Y. Li	Mechanical Engineer	Mechanical Engineering
L. Reiter	Leader, Seismology Section	Geosciences

APPENDIX J

REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 6, 1984

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE LIMERICK GENERATING STATION

The Committee commented on the application for a permit to construct this Station in a report dated August 10, 1971, and on the application to operate this Station in an interim report dated October 18, 1983. During its 295th meeting, November 1-3, 1984, the Advisory Committee on Reactor Safeguards continued its review of the application of the Philadelphia Electric Company (Applicant) for a license to operate the Limerick Generating Station, Units 1 and 2. This phase of the review was continued during ACRS Subcommittee meetings held on October 9-10 and October 20, 1984. The Committee also had the benefit of the documents referenced. During its review, the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff as well as written and oral statements from members of the public.

The Committee stated in its October 18, 1983 interim report that, because of the uncertain schedule for Unit 2, it was not appropriate to report on Unit 2 at that time. Construction on Unit 2 has been stopped, but may be resumed after the start of operation of Unit 1. We do not believe it is appropriate for the Committee to report on Unit 2 at this time.

The Committee in its October 18, 1983 report stated that it had not completed its review and listed a number of matters yet to be considered. These matters have been discussed at subsequent Subcommittee and Committee meetings, and we conclude that they have been dealt with satisfactorily.

In response to a request from the NRC Staff, the Applicant submitted a probabilistic risk assessment (PRA) in March 1981. A supplement to this PRA report was submitted in April 1983 in the form of a severe accident risk assessment (SARA) report. The NRC Staff has reviewed this study and has used results from this study in the Environmental Statement for this Station. The Applicant has used insights from this PRA/SARA evaluation in the design of and in the development of operational procedures for the Limerick plant. The Applicant, in discussions with the Committee, demonstrated an understanding of the methodology and its uses and a commitment to its application in the operation of the Limerick plant. The Applicant is to be commended for this work.

The Limerick PRA/SARA includes a seismic risk analysis which reflects the state-of-the-art which was used in the Zion and Indian Point PRAs. The results obtained by these methods are characterized by large uncertainties, and are the subject of disagreement in the scientific and engineering communities. We believe that the NRC and the industry should continue to work to develop methods which can be used to quantify seismic risk and to identify any seismic outliers which might exist.

The Committee has previously recommended that the Zion and Indian Point plants be reviewed for systems interactions that might lead to significant degradation of safety. The issue of systems interactions is currently being addressed under the USI A-17, "Systems Interactions in Nuclear Power Plants." Philadelphia Electric Company has already examined many of the possible systems interactions in the Limerick plant. However, in view of the demography of the site, we recommend that Limerick receive special attention in the NRC Staff's consideration of USI A-17.

We believe that, subject to the resolution of open items identified by the NRC Staff and subject to the satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Limerick Generating Station, Unit 1 can be operated at power levels up to 3293 MWt without undue risk to the health and safety of the public.

Additional comments by ACRS Member David Okrent are presented below.

Sincerely,

David A. Ward

David A. Ward
Acting Chairman

Additional Comments by ACRS Member David Okrent

The matter of potential improvements in design either to prevent or to mitigate severe accidents received only limited attention by the NRC Staff during this review. Further studies are in progress which should be completed and evaluated in the next two or three years. At that time, the Limerick Generating Station should be reviewed for the possible desirability and appropriateness of such improvements.

References:

1. Philadelphia Electric Company, "Final Safety Analysis Report, Limerick Generating Station, Units 1 and 2," Revisions 21-36
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Limerick Generating Station, Units 1 and 2," Supplement No. 1, USNRC Report NUREG-0991, dated December 1983

3. BNL Report Prepared for U. S. Nuclear Regulatory Commission, "A Review of the Limerick Generating Station Severe Accident Risk Assessment" - Review of Core Melt Frequency, NUREG/CR-3493 and BNL-NUREG-51711, dated July 1984
4. BNL Report Prepared for U. S. Nuclear Regulatory Commission, "A Review of the Limerick Generating Station Probabilistic Risk Assessment," NUREG/CR-3028 and BNL-NUREG-51600, dated February 1983
5. U. S. Nuclear Regulatory Commission, "Review Insights on the Probabilistic Risk Assessment for the Limerick Generating Station," USNRC Report NUREG-1068, dated August 1984
6. Letter from A. Schwencer, NRC Division of Licensing, to Edward G. Bauer, Jr., Philadelphia Electric Company, Subject: Review of Limerick Severe Accident Risk Assessment, dated June 22, 1984, with attachment, BNL-33835, "Containment Failure Mode and Fission Product Release Analysis for the Limerick Generating Station: Base Case Assessment"
7. Letter from M. Lewis, Member of the Public, to R. Savio, Advisory Committee on Reactor Safeguards, regarding the ACRS review of the Limerick Generating Station, dated October 3, 1984

APPENDIX P

TECHNICAL EVALUATION REPORT
LIMERICK GENERATING STATION
CONFORMANCE TO REGULATORY GUIDE 1.97

CONFORMANCE TO REGULATORY GUIDE 1.97
LIMERICK GENERATING STATION UNIT NOS. 1 AND 2

R. VanderBeek

Published October 1984

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
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ABSTRACT

This EG&G Idaho, Inc., report provides a review of the submittal for the Limerick Generating Station Unit Nos. 1 and 2, and identifies areas of full conformance to Regulatory Guide 1.97, Revision 2. Any exception to these guidelines are evaluated and those areas where sufficient basis for acceptability is not provided are identified.

FOREWORD

This report is supplied as part of the "Program for Evaluating Licensee/Applicant Conformance to RG 1.97," being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Systems Integration, by EG&G Idaho, Inc., NRC Licensing Support Section.

The U.S. Nuclear Regulatory Commission funded the work under authorization B&R 20-19-40-41-3.

Docket Nos. 50-352 and 50-353

CONTENTS

CONFORMANCE TO REGULATORY GUIDE 1.97
LIMERICK GENERATING STATION UNIT NOS. 1 AND 2

1. INTRODUCTION

On December 17, 1982, Generic Letter No. 82-33 (Reference 1) was issued by D. G. Eisenhut, Director of the Division of Licensing, Nuclear Reactor Regulation, to all licensees of operating reactors, applicants for operating licenses and holders of construction permits. This letter included additional clarification regarding Regulatory Guide 1.97, Revision 2 (Reference 2), relating to the requirements for emergency response capability. These requirements have been published as Supplement 1 to NUREG-0737, "TMI Action Plan Requirements" (Reference 3).

The Philadelphia Electric Company, applicant for the Limerick Generating Station, Unit Nos. 1 and 2, provided a response to the generic letter on April 15, 1983 (Reference 4). This submittal refers to the Limerick Final Safety Analysis Report (FSAR, Reference 5) for the review of the instrumentation provided in conformance to Regulatory Guide 1.97. Additional information was submitted August 16, 1984 (Reference 7).

This report provides an evaluation of the submittals and the referenced FSAR.

2. REVIEW REQUIREMENTS

Section 6.2 of NUREG-0737, Supplement 1, sets forth the documentation to be submitted in a report to NRC describing how the applicant meets the guidance of Regulatory Guide 1.97 as applied to emergency response facilities. The submittal should include documentation that provides the following information for each variable shown in the applicable table of Regulatory Guide 1.97.

1. Instrument range

2. Environmental qualification

3. Seismic qualification

4. Quality assurance

5. Redundance and sensor location

6. Power supply

7. Location of display

8. Schedule of installation or upgrade.

Further, the submittal should identify deviations from the guidance in the Regulatory Guide and provide supporting justification or alternatives.

Subsequent to the issuance of the generic letter, the NRC held regional meetings in February and March 1983, to answer licensee and applicant questions and concerns regarding the NRC policy on this matter. At these meetings, it was noted that the NRC review would only address exceptions taken to the guidance of Regulatory Guide 1.97. Further, where licensees or applicants explicitly state that instrument systems conform to

the provisions of the guide, it was noted that no further staff review would be necessary. Therefore, this report only addresses exceptions to the guidance of Regulatory Guide 1.97. The following evaluation is an audit of the applicant's submittal based on the review policy described in the NRC regional meetings.

3. EVALUATION

In the applicant's response to NRC Generic Letter 82-33, Section 7.5 of the Limerick FSAR is identified as containing (a) the description of the Post-Accident Monitoring System (PAMS), (b) tables which identify the monitored parameters, and (c) compliance to, or deviations from, the guidance of Regulatory Guide 1.97 along with the supporting justification or alternatives. This evaluation is based on the information provided in Section 7.5 of the Limerick FSAR and the additional information provided in the August 16, 1984 (Reference 7) submittal.

3.1 Adherence to Regulatory Guide 1.97

The applicant has stated in Section 1.8 of the FSAR that Limerick is in conformance to Regulatory Guide 1.97 to the extent discussed in Section 7.5 of the FSAR. Within Table 7.5.3 of the Limerick FSAR, the applicant has identified the post-accident monitoring instrumentation that provides indication of Regulatory Guide 1.97 variables. The applicant has made an explicit commitment to conform to the guidelines of Regulatory Guide 1.97 with the exception of the identified deviations noted in Section 3.3 of this report.

3.2 Type A Variables

Regulatory Guide 1.97 does not specifically identify Type A variables, i.e., those variables that provide information required for operator controlled safety actions. The applicant has classified the following instrumentation channels as Type A variables:

1. Primary containment oxygen and hydrogen concentration

2. Reactor pressure vessel pressure

3. Reactor pressure vessel water level

4. Suppression pool water temperature

5. Suppression pool water level

6. Drywell pressure.

The above variables are included as Type B, C, D or E variables. All of the above variables are identified by the applicant as conforming to Regulatory Guide 1.97.

3.3 Exceptions to Regulatory Guide 1.97

The licensee identified the following exceptions to the requirements of Regulatory Guide 1.97.

3.3.1 Neutron Flux

Exception has been taken by the applicant to the Category 1 recommendation of Regulatory Guide 1.97 for the neutron flux measurement. Category 2 drive mechanisms for the start up range detectors, and the reactor protection system power supplies are used. All other components are Category 1. The applicant indicates that the only event requiring the long term surveillance of neutron flux is an anticipated transient without scram (ATWS), and any decision to upgrade depends on the resolution of the ATWS issue. The applicant states that there are 4 source range monitors, 8 intermediate range monitors and 6 average power range monitors. As there is sufficient redundancy of instrumentation and there is less importance to safety for the ATWS issue, the applicant considers the instrumentation acceptable and consistent with any future ATWS rulemaking.

We concur with the applicant, and find the applicant's position acceptable.

3.3.2 Reactor Water Level

Exception has been taken by the applicant to Regulatory Guide 1.97 for the reactor water level measurement. The guide specifies that the range should be from the bottom of the core support plate to the lesser of the top of vessel or the centerline of main steamlines. The range provided by the applicant for the upper limit of water level is 5 ft. less than that recommended by Regulatory Guide 1.97. The applicant states that the need for the upper range specified in Regulatory Guide 1.97 does not exist. The justification is that the applicant has provided two (2) additional level 8 trips in both the HPCI and RCIC trip circuits. Therefore, the present design is sufficient to (a) determine the reactor water level over the range required by the operator for normal and emergency operation, (b) determine the adequacy of core cooling, and (c) limit reactor level to level 8.

We concur with the applicant that with the installation of the two additional level 8 HPCI and RCIC trip signals, that it is not necessary to measure reactor water level to the centerline of the main steamlines. We find the deviation acceptable.

3.3.3 Drywell Sump Level and Drywell Drain Sumps Level

Exception has been taken by the applicant to Regulatory Guide 1.97 for the Drywell Sumps Level and the Drywell Drain Sumps Level measurements. The applicant position is that the Drywell Sump Level and Drywell Drain Sumps Level instrumentation should be qualified to Category 3 instead of Category 1 requirements. The supporting justification is that (a) the drywell pressure and temperature along with the primary containment area radiation can be used to provide indication of leakage in the drywell, (b) these variables are qualified to Category 1 or 2, (c) the drywell sump systems are isolated for accident conditions, (d) a sump tank level monitoring system that meets the requirements of Regulatory Guide 1.45 has been installed.

We find the sump tank level monitoring system that is specified in Regulatory Guide 1.45 a method of determining leakage from the reactor coolant system. Once the drywell sump systems are isolated for accident conditions, no useful post-accident information would be available and the operator is able to tell that the tank is full by using the existing instrumentation.

Based on the above, the applicant's deviation from Category 1 to Category 3 instrumentation is acceptable.

3.3.4 Radioactivity Concentration or Radiation Level in Circulating Primary Coolant

Exception has been taken by the applicant to Regulatory Guide 1.97 for this measurement. A Category 3 classification has been assigned to this variable instead of the recommended Category 1 classification per Regulatory Guide 1.97. The applicant indicates that radiation level measurements to indicate fuel cladding failure are provided by the following instrumentation

1. Post-Accident Sampling System (PASS)

2. Condenser off-gas radiation monitors

3. Main steam line radiation monitors

4. Primary containment radiation monitors

5. Containment hydrogen concentration monitors.

We concur with the justification submitted by the applicant for this deviation. Their existing instrumentation is adequate to monitor post-accident reactor coolant activity. Further, a continuous post-accident reactor coolant activity monitor is not a requirement of NUREG-0737. Therefore, this is an acceptable deviation from Regulatory Guide 1.97.

3.3.5 Radiation Exposure Rate

The applicant has elected not to implement this type C variable as recommended by Revision 2 of Regulatory Guide 1.97, the justification being that the applicant feels other means, such as noble gas monitoring, are better suited for breach detection. The applicant states that exposure rate monitors are being provided for indication of habitability only. Revision 3 of Regulatory Guide 1.97 (Reference 6) deletes this Type C variable from the recommended instrumentation. Therefore, lack of the instrumentation for this variable is acceptable.

3.3.6 Suppression Spray Flow and Drywell Spray Flow

The applicant has chosen to provide an alternate means for monitoring the suppression spray flow and drywell spray flow measurements. The applicant has stated that these variables are accurately and reliably monitored by the Residual Heat Removal System loop flow and valve position indication. The RHR loop flow indicator provides drywell spray and suppression flow indication. The valve position indicators allow the operator to verify that drywell spray and suppression pool spray flows are directed through the proper flow paths. Also, suppression pool air space temperature and pressure and drywell temperature indicators provide direct and unambiguous indication of the effectiveness of drywell spray and suppression pool spray. This instrumentation meets the requirements of Regulatory Guide 1.97, Rev. 2, and is sufficient to satisfy the operator's information requirements for the drywell and suppression pool sprays as called out by Limerick procedures.

We concur with the applicant that the above described instrumentation is sufficient.

3.3.7 Standby Liquid Control System (SLCS) Flow

Exception has been taken by the applicant to Regulatory Guide 1.97 for the SLCS flow measurement. The applicant states that the SLCS flow can be adequately monitored by (a) the decrease in the level of the boric acid

storage tank, (b) the reactivity change in the reactor as measured by neutron flux and concentration of boron, (c) the SLC pump motor contactor indicating lights (or motor current) or (d) the squib valve continuity indicating lights.

Based on the above justification, we find that the applicant's position meets the requirements of Regulatory Guide 1.97 for this variable.

3.3.8 Reactor Building or Secondary Containment Area Radiation

Exception has been taken by the applicant to Regulatory Guide 1.97 for the Reactor Building or Secondary Containment Radiation measurement. Regulatory Guide 1.97 recommends that this variable be monitored, whereas the applicant's position is that the secondary containment area radiation is not an appropriate parameter to use to detect or assess primary containment leakage. Therefore, the applicant deems that reactor enclosure area radiation monitors are not required for the Limerick secondary containment.

This exception goes beyond the scope of this review; however, the information provided by the applicant has been reviewed and approved by the NRC Radiological Assessment Branch.

We concur with this review.

3.3.9 Radiation Exposure Rate

Exception has been taken by the applicant to Regulatory Guide 1.97 for the radiation exposure rate Type E measurement. Regulatory Guide 1.97 recommends that this variable be monitored, whereas the applicant's position is (a) the Limerick design does not require access to harsh environment area to service safety-related equipment during an accident and (b) portable radiation monitors will be provided to establish accessibility.

This exception goes beyond the scope of this review; however, the information provided by the applicant has been reviewed and approved by the NRC Radiological Assessment Branch.

We concur with this review.

3.3.10 Plant and Environs Radiation

Regulatory Guide 1.97 specifies that plant and environs radiation should be monitored over the range of 10^{-3} R/hr to 10^{4} R/hr, photons and 10^{-3} Rads/hr to 10^{4} Rads/hr, beta radiation and low energy photons. The classification is Category 3.

The applicant has identified within Section 12.5.2.2.3 of the FSAR the instrumentation that will be used to monitor the Plant and Environs Radiation along with additional equipment to enhance post-accident monitoring. The post-accident instrumentation will be comprised of low, medium, and high range portable ion chambers (1 mR/hr to 20,000 R/hr gamma and 20,000 Rad/hr beta), open window alpha scintillation probes, Geiger Mueller (GM) probes, energy compensated beta/gamma GM probes (for low energy photons), and portable beta/gamma geiger counters. Audio speakers, alarming count rate meters, and extension arms will be provided for attachments to the survey instruments. Airborne radioactivity levels will be determined from laboratory analysis of particulate filters and iodine cartridge samples obtained with high and low volume samplers. Portable instruments and equipment reserved for emergency use will be located at an assembly area remote from the main plant.

It is concluded that the applicant meets the intention and purpose of the Regulatory Guide recommendations and the alternate instrumentation is acceptable.

3.3.11 Accident Sampling (Primary Coolant, Containment Air and Sump)

Exception has been taken by the applicant to Regulatory Guide 1.97 for the Primary Coolant and Sump Grab Sampling variables. Regulatory Guide 1.97 recommends that the primary coolant and sump be sampled, whereas the applicant's position is that sampling of the suppression pool is representative of the variables.

The applicant takes exception to the guidance of Regulatory Guide 1.97 with respect to post accident sampling capability. This exception goes beyond the scope of this review; however, sampling of the suppression pool in lieu of the primary coolant and sumps has been reviewed and approved by the NRC Materials and Chemical Engineering Branch as part of their review of NUREG-0737, Item II.B.3 as referred to in Reference 7.

4. CONCLUSIONS

With the exceptions identified by the applicant in Section 7.5.2.5 of the FSAR, the applicant has committed conformance to Regulatory Guide 1.97. Review of the applicant's justification for taking the identified exceptions to the regulatory guide concludes that the exceptions are acceptable.

5. REFERENCES

1. NRC letter D. G. Eisenhut, to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits, "Supplement No. 1 to NUREG-0737--Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

2. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 2, U.S. Nuclear Regulatory Commission (NRC), Office of Standards Development, December 1980.

3. Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability, NUREG-0737 Supplement No. 1, NRC, Office of Nuclear Reactor Regulation, January 1983.

4. Philadelphia Electric Company Letter to NRC, V. S. Boyer to Darrell Eisenhut, Director, Division of Licensing, "Limerick Generating Station, Units 1 and 2, Docket Nos. 50-352 and 50-353," April 15, 1983.

5. Final Safety Analysis Report (FSAR) for Limerick Generating Station Unit 1 and Unit 2, Revision 25, May 1983.

6. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 3, U.S. Nuclear Regulatory Commission (NRC), Office of Standards Development, May 1983.

7. Philadelphia Electric Company Letter to NRC, J. S. Kemper to A. Schwencer, Chief, Division of Licensing, "Limerick Generating Station, Units 1 and 2,Conformance to Regulatory Guide 1.97," August 16, 1984.

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13. ABSTRACT (200 words or less)

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

Supplement 1 was issued in December 1983 and addressed several outstanding issues. It 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 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. This Supplement 4 addresses some of those technical issues and their associated license conditions which require resolution prior to proceeding beyond the five percent power level. The remaining issues will be addressed in a later supplement to this report. This Supplement 4 also contains 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.

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