NUREG-0991
Supplement No. 7

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

April 1989

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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 licensee) 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. Supplements 2 and 3 were issued in October 1984. License NPF-27 for the low-power operation of Limerick Unit 1 was issued on October 26, 1984. Supplement 4 was issued in May 1985, Supplement 5 was issued in July 1985, and Supplement 6 was issued in August 1985. These supplements addressed further issues that required resolution before Unit 1 proceeded beyond the 5-percent power level. The full-power operating license for Limerick Unit 1 (NPF-39) was issued August 8, 1985, and the unit has completed two cycles of operation.

This supplement addresses the major design differences between Units 1 and 2, the resolution of all issues that remained open when the Unit 1 full-power license was issued, the staff's assessment of the licensee's application to operate Unit 2, and the issues that require resolution before issuance of an operating license for Unit 2.

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APPENDICES

1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

In August 1983, the Nuclear Regulatory Commission (hereinafter referred to as the NRC or the staff) issued its Safety Evaluation Report (SER), NUREG-0991, regarding the application by the Philadelphia Electric Company (hereinafter referred to as PECo or the licensee) for licenses to operate the Limerick Generating Station, Units 1 and 2, Docket Nos. 50-352 and 50-353. Supplement 1 to the SER was issued in December 1983, Supplements 2 and 3 were issued in October 1984, and Operating License NPF-27, authorizing power up to 5 percent, was issued on October 26, 1984. Supplement 4 to the SER was issued in May 1985, Supplement 5 was issued in July 1985, and Supplement 6 was issued in August 1985. These supplements addressed issues that required further resolution before Unit 1 proceeded beyond the 5-percent power level. A full-power operating license (NPF-39) was issued for Limerick Unit 1 on August 8, 1985.

As noted above, the staff's SER assessed operation of both Limerick Units 1 and 2. Construction of Unit 2 was halted in January 1984 by Order of the Pennsylvania Public Utility Commission. At the time, construction was about 30 percent complete. Construction of Unit 2 resumed in February 1986 with PECo's agreement to accept a cost containment cap of about $3.1 billion for construction and certain operational incentive programs. On May 3, 1988, the Commission modified Construction Permit CPPR-107 to extend the earliest and latest completion dates to May 1, 1989, and January 1, 1992, respectively.

This document, the seventh supplement to the SER (SSER 7), primarily relates to Unit 2 and provides additional information supporting the issuance of a low-power operating license for fuel loading, initial criticality, and power ascension up to 5 percent for Limerick Unit 2.

Each of the sections and appendices of this supplement is designated the same as the related portion of the SER. Each section is supplementary to and not in lieu of the discussion in the SER and Supplements 1 through 6, unless otherwise noted. Appendix A is a continuation of the chronology of this safety review. Appendix H lists the principal contributors. Appendix Q provides the evaluation of the preservice inspection relief requests for Unit 2. Appendix R is an updated staff response to the ACRS report of November 6, 1984, on the Limerick Generating Station.

Copies of this supplement are available for inspection at the NRC Public Document Room, 2120 L Street N.W., Washington, D.C. and at the local Public Document Room at the Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

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Office of Nuclear Reactor Regulation
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1.8 Outstanding Issues

The only remaining outstanding issue in SSER 6 was the conduct of a second full-participation emergency preparedness exercise. A full-participation exercise was satisfactorily held April 3, 1986. There are no outstanding issues.

1.10 License Condition Items

When the full-power license was issued for Unit 1 on August 8, 1985, there were 11 license conditions that had to be completed before startup following the first refueling outage. All of the required actions have been verified to have been satisfactorily completed, including adding appropriate surveillance requirements in the Technical Specifications, when applicable.

1.12 Modifications to the Facility Subsequent to Unit 1 Licensing

There have been a number of design changes and modifications made to Limerick Unit 1 since its license was issued on August 8, 1985. Some of these changes are the licensee's initiative, some are required by the NRC, and some required as a condition of the license. All design changes and modifications approved by the staff for Unit 1 also have been implemented for Unit 2. The more significant modifications that required staff approval are listed below.

Modification	Required/ Initiated By	SSER 7 Section
Anticipated Transient Without Scram (ATWS) Rule	10 CFR 50.62	7.2.2.5
Connection of Standby Gas Treatment System (SGTS) to Refueling Floor	L.C.2.C.14	6.2.3
Install Isolation Valve in Hydrogen Recombiner Line	L.C.2.C.11	Not addressed
Remote Shutdown System Redundancy	L.C.2.C.12	Not addressed
Reactor Enclosure Cooling Water Isolation Valves	L.C.2.C.10	Not addressed
Add Stairway	L.C.2.C.3	Not addressed
Elimination of Reactor Enclosure Recirculation System Cooldown Mode	Licensee	Not addressed
Eliminate Reactor Sequence Control System/Lower Rod Worth Minimizer Set Point	Licensee	Not addressed
Increase SGTS Capacity	Licensee	6.2.3

1.13 Differences in Design Between Units 1 and 2

In addition, there are two significant differences in design between Units 1 and 2 that have been approved by the staff. On March 27, 1986, the licensee requested approval for use of American Society of Mechanical Engineers (ASME) Code Cases N-397 and N-411 for Unit 2 as part of a snubber optimization program. By letter dated April 25, 1986, the licensee requested approval to eliminate the use of the arbitrary intermediate break assumptions in pipe rupture analysis for Unit 2. These two design changes are discussed in Sections 3.7 and 3.6.2.1, respectively. The licensee also informed the staff that Unit 2 will be using an enhanced safety parameter display system. This is discussed in Section 18.2 of this supplement.

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 Postulated Rupture of Piping

3.6.2.1 Elimination of Arbitrary Intermediate Pipe Breaks

By letter dated April 25, 1986, the licensee proposed to eliminate postulated arbitrary intermediate pipe breaks (AIPB) from the design analyses of Limerick Unit 2.

Since November 1984 the NRC staff has approved requests from several utilities to eliminate AIPB analysis. In its approval for these specific plants, the staff stated that the elimination of AIPB analysis was not to apply to piping systems in which stress corrosion cracking, large unanticipated dynamic loads such as steam or water hammer, or thermal fatigue in fluid mixing situations could be expected to occur. However, in Volume 3 of NUREG-1061 ("Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," November 1984), the staff states that after additional review, it is realized that in certain systems and for certain materials, thermal fatigue and stress corrosion cracking cannot be absolutely excluded from piping operation, nor can steam or water hammer. It also may never be possible to specify precise acceptable levels of thermal fatigue and stress corrosion cracking, or to ensure analytically that these levels would not be exceeded. If these unanticipated severe conditions were to occur, the break would most likely be located at the terminals ends, at the connections to components, and at other locations that introduce higher stress concentrations or that exceed the stated threshold limits in Standard Review Plan (SRP, NUREG-0800) Section 3.6.2. Therefore, these locations would not be affected by relaxing the requirement to postulate AIPBs. In Volume 5 of NUREG-1061 (November 1985), the staff states that pipe rupture protective devices can introduce many negative effects on plant operation and do not contribute to plant safety as originally intended. Therefore, deletion of the requirement to postulate AIPBs and the need to protect against dynamic effects such as jet impingement and pipe whip is warranted.

It should be noted that the staff's approval to eliminate AIPB analysis applies to pipe rupture protective devices only. For environmental qualification of equipment and structural design of compartments or enclosures traversed by high-energy piping systems, breaks will continue to be postulated in accordance with the present Limerick project criteria; that is, in each compartment or enclosure traversed by a high-energy piping system, nonmechanistic breaks are postulated to establish environmental consequences. Therefore, elimination of the AIPB analysis will not change the existing environmental or structural criteria for any structure, system, or component. The staff concludes that the licensee's request to eliminate AIPB design analysis, as presented in the referenced letter, is justified and acceptable.

This information was previously transmitted to the licensee on June 27, 1986. The elimination of AIPB design analysis is one of the two more significant differences in design between Limerick Units 1 and 2.

3.7 Seismic Design

By letter dated March 27, 1986, the licensee requested authorization to use the damping values of ASME Code Case N-411, "Alternative Damping Values for Seismic Analysis of Classes 1, 2, and 3 Piping Sections, Section III, Division I," in lieu of the damping values identified in the Limerick Unit 2 Final Safety Analysis Report (FSAR). The staff responded to that request by letter dated May 30, 1986. That letter provided details of the staff's position on the applicability and condition for acceptable use of ASME Code Case N-411. One of the conditions identified was that damping values from Code Case N-411 may be used only in piping analyses for which enveloped response spectra are used. After reviewing the staff's position, the licensee informed the staff that Code Case N-411 damping values had been used for piping system analyses using the independent support motion (ISM) response spectrum method.

After this inconsistency in the analyses was identified, meetings between PECo and the staff were held in Bethesda, Maryland, on October 8, November 7, and December 3, 1986 and January 15, 1987. During these meetings, the licensee presented its various analytical approaches with N-411 damping values (ISM/N-411) for Limerick Unit 2 piping. Details of the analyses were subsequently submitted to the staff by letters dated December 24, 1986, and January 13 and 26, 1987.

These submittals provide details relative to various approaches to justify the use of the ISM/N-411 method for Limerick Unit 2 piping systems. During the meeting of October 8, 1986, the licensee indicated that there are 31 piping systems (6 within the General Electric (GE) scope of supply and 25 within the Bechtel scope of supply) analyzed using the ISM/N-411 method. For the six nuclear steam supply system (NSSS) piping systems, two time-history analyses were performed (recirculating loop A and main steam line D) using FSAR damping values. A comparison of the results of those two time-history analyses with the results of the ISM (square root sum of the squares [SRSS])/N-411 method showed that both methods gave similar predictions of piping stresses and restraint loads at similar locations and that adequate margins with respect to piping allowable stresses and support design loads remained.

In its submittal of December 24, 1986, the licensee further demonstrated similarities of main steam lines A, B, and C with main steam line D, as well as similarities of recirculation loop B with recirculation loop A. The similarities considered included piping layout, support location, natural frequencies of piping stress, and restraint loads. On the basis of the demonstrated similarities, the licensee concluded that the design of the four NSSS piping systems, which have not been specifically reanalyzed by the time-history methodology, are acceptable. On the basis of the information presented by the licensee, the staff agrees with the licensee's conclusion.

For the 25 balance-of-plant (BOP) piping systems using the ISM (absolute sum)/N-411 method, 3 piping systems (feedwater line A, residual heat removal [RHR] return line A, and main steam line D safety relief valve [SRV] discharge lines) were reanalyzed by the time-history methodology and were shown to be acceptable by a comparison of the results of ISM (absolute sum)/N-411 method with the results of the time-history analyses.

The licensee then further extended the three BOP time-history evaluations to the acceptability of feedwater line B, RHR return line B, and all other main steam SRV discharge lines on the basis of their respective similarities. On January 14, 1987, the licensee also indicated that there are three piping systems for which the ISM/N-411 method were utilized only for the annulus pressurization (AP) evaluation. For these piping systems, the majority of modes are excited at frequencies greater than 20 Hz where the N-411 damping valve has no effect because the Regulatory Guide (RG) 1.61 damping value is greater than or equal to the N-411 damping value above 20 Hz. Nevertheless, the effects of AP loading for frequencies below 20 Hz were assessed and were determined to be acceptable because adequate margins relative to the design allowables remained. For the remaining BOP piping systems using the ISM/N-411 method, a study calculation approach was used. Study calculations were performed using either ISM (absolute sum) with RG 1.61 damping values or enveloped response spectra with N-411 damping values. On January 26, 1987, the licensee stated that all study calculations had been completed and the results were determined to be satisfactory because pipe stresses were below Code allowables and no restraint modifications were required. The licensee further stated that it had reviewed four study calculations that had the most significant increases in maximum pipe stresses between the study calculation and the existing record calculation for the effect on nozzle loads and valve accelerations and had determined that they were not adversely affected by using the ISM (absolute sum)/N-411 method. Based on the results of its assessment as described above, the licensee concluded that it had adequately demonstrated the acceptability of those 25 BOP piping systems.

On the basis of its review of the licensee's evaluation as described above, the staff concludes that the licensee has provided acceptable justification for the design of these 31 piping systems using the ISM/N-411 method for its analyses. The analyses performed, using the ISM/N-411 method, may remain the calculations of record. It should be noted that use of the ISM/N-411 method will be limited to those cases only because they were shown to be acceptable by studies that compared results with results of analyses performed using currently acceptable methods. Any further piping design, or analyses/reanalyses of piping systems other than those identified above, or reanalyses of these 31 piping systems because of a significant change in the routing of the piping and/or pipe support modification should be performed using one of the following analytical methods approved by the staff:

- time-history analysis with FSAR damping values
- enveloped response spectra analysis with ASME Code Case N-411 damping values
- ISM (absolute sum) response spectra analysis with RG 1.61 damping values

The above information was previously transmitted to the licensee on March 27, 1987.

3.9 Mechanical Systems and Components

3.9.6 Inservice Testing of Pumps and Valves

Following meetings and discussions with the staff, the licensee submitted (November 23, 1988) a revised pump and valve inservice testing (IST) program for Unit 1 for the first 10-year interval which commenced February 1, 1986 (the date of commercial operation). This revision also serves as the submittal of the IST Program for Unit 2 when it begins its first 10-year interval. The NRC staff has not completed its review of the licensee's program, including cold shutdown test justifications and pump and valve relief requests. However, on the basis of its review of the information in the licensee's program and on flow diagrams in the Limerick FSAR, the NRC staff finds that it is impractical within the limitations of design, geometry, and accessibility for the licensee to meet some of the requirements of the ASME Code. Imposition of those requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality or safety. Therefore, pursuant to 10 CFR 50.55a, the staff concludes that the relief the licensee has requested from the pump and valve testing requirements of 10 CFR 50.55(g)(2) and (g)(4)(i) is warranted for a portion of the initial 120-month period, during which the NRC staff expects to complete its detailed review. This interim relief will terminate once the staff's final safety evaluation on the IST program is issued.

4 REACTOR

4.2 Fuel System Design

4.2.3 Design Evaluation

4.2.3.1 Fuel System Damage Evaluation

(4) External Corrosion and Crud Buildup

In the SER the staff discussed the potential for what General Electric (GE) has characterized as "crud-induced localized corrosion" (CILC) failures based on the information available in late 1982. The staff noted that the corrosion is associated with a high copper concentration in the core coolant water and Zircaloy cladding metallurgy susceptible to nodular corrosion. The staff also noted that there was not much substantive information regarding (a) the corrosion mechanism, (b) the identification of susceptible material, (c) the initial or threshold concentration of copper in the reactor water that is likely to cause the nodular corrosion, and (d) the metallurgical feature that determines the susceptibility of Zircaloy tubing to CILC.

As of December 3, 1987, over 3.3 million GE 8x8 fuel type production Zircaloy-clad UO_2 fuel rods were in, or had completed, operation in commercial boiling-water reactors (BWRs). As of the same date, over 1.5 million GE fuel rods were in operation. The performance of GE 8x8 fuel type Zircaloy-clad fuel rods continues to be highly successful as demonstrated by a 1987 fuel rod reliability rate of greater than 99.99 percent. (Letter from J. S. Charnley [GE] to M. W. Hodges [NRC] dated November 16, 1988, forwarding report on "Experience with BWR Fuel Through December 1987.")

Early GE fuel operating experience identified fuel performance problems that have been subsequently corrected through evolutionary design and manufacturing and operating experience. These earlier problems are not affecting fuel performance at this time. In the above referenced report, GE stated that pellet-cladding interaction (PCI) and CILC are the only cladding perforation mechanisms that have affected fuel performance in recent years. In 1983, GE commercially introduced zirconium barrier fuel as a material solution to the PCI problem. Operational procedures, which involve slow approaches to power, had essentially eliminated PCI failures in light-water reactors (LWRs). The staff's evaluation of fuel performance indicates that the combination of fuel conditioning and use of zirconium barrier fuel has eliminated PCI as a potential fuel-cladding failure mechanism. (See NUREG/CR-3950, Volume 5, "Fuel Performance Annual Report for 1987," December 1988.) The fuel inserted in the first Limerick Unit 1 reload, the fuel purchased for the second Unit 1 reload, and all of the fuel procured for Unit 2 is the new GE 8x8 barrier fuel. PCI is not expected to be a problem with Limerick Unit 2. CILC, however, is still a potential cause of fuel-cladding failures in some types of BWRs such as Limerick Units 1 and 2.

The first incident of CILC occurred at Vermont Yankee during 1978-1979, followed by similar incidents at Hatch Units 1 and 2, Browns Ferry Units 1 and 2, Peach Bottom Units 2 and 3, and Ringhals 1 (Sweden) during 1979-1987. All of these plants are GE BWR/4 type NSSS except for Ringhals 1, which is an ASEA-Atom (architect/engineeer) NSSS. The majority of fuel rods in the GE plants that were identified as failed contained UO_2-Gd_2O_3 burnable absorber. The failed rods in Ringhals 1 were primarily UO_2. In all cases, the failed rods were 8x8 type fuel rods. (See Electric Power Research Institute [EPRI] Report NP-3789," Corrosion Product Buildup on LWR Fuel Rods, April 1985.)

The power density in the BWR/4 types, which have experienced CILC, is higher than older plants. The BWR/4 power density is about 51 kW/l compared with 46 kW/l for Big Rock Point, 41 kW/l for Ringhals 1 and KRB-A (Germany), 31 kW/l for Dresden-1, and 11 kW/l for SGHWR (United Kingdom). The linear power of the 8x8 BWR/4 and Ringhals 1 fuel rods is comparable to fuel rods in the older plants with 7x7 equivalent fuel rod diameters (see EPRI NP-3789).

There was an indication of a fuel leak in Limerick Unit 1 in March 1988. The leak was immediately detected by the increase in off-gas activity. Power was reduced to limit off-gas activity to a self-imposed level that was a small percentage of the activity limits of the Technical Specifications. Throughout the summer and fall of 1988, there was evidence of additional leaking fuel pins and power was incrementally reduced. At the time that Unit 1 was shut down for the second refueling on January 11, 1989, the unit was only operating at 41 percent power. By reducing power, the licensee limited the average off-gas activity to about 10 percent of the regulatory limit.

Limerick Unit 1 was shut down on January 11, 1989, for the second refueling outage; Cycle 3 restart was projected for April 1989. The licensee had planned to replace 224 fuel assemblies, which is in the range of a normal reload. On the basis of the estimated number of leaking fuel pins, the licensee projected in the fall of 1988 that it also would have to utilize 40 of the new Unit 2 fuel assemblies that were on site and reinsert a number of reconstituted and non-reconstituted fuel assemblies from the spent fuel pool that had been discharged during the first refueling. In the fall of 1988, the licensee inspected the fuel assemblies in the pool that were the most likely candidates for reinsertion. The inspection disclosed that although there had been no perforation of the cladding, a number of fuel pins in some assemblies had externally corroded cladding that varied from typical nodular corrosion to gross corrosion where the surface was flaking off. The licensee proposed to reconstitute 84 of the initial fuel assemblies, replacing non-heat-treated pins, and selected an additional 148 clean or minimally oxidized bundles for reinsertion. The bundles with the most severe corrosion were primarily those that had experienced the higher heat fluxes and were predominantly UO_2-Gd_2O_3 rods. By letter dated January 27, 1989, PECo submitted an application for an amendment to change the Technical Specifications to accommodate the second refueling of Unit 1. The letter described the proposed mix of new, previously irradiated, and reconstituted fuel bundles on the basis of what was known of the status of the fuel in the reactor at the time.

Following the January 1989 shutdown, initial sipping of 296 fuel assemblies from the reactor disclosed 5 leaking fuel assemblies from the initial core load and 13 leaking bundles from the fuel assemblies of the first Unit 1 reload (reload-1) that had only operated for one fuel cycle. An inspection was performed on some of the off-loaded fuel assemblies, using an underwater periscope and video camera. In 26 bundles that were examined from the initial core, about 12 percent of the pins were classified as visual standard 4, 5, or 6.* Inspection of the reload-1 bundles that had only been exposed for one cycle revealed that two-thirds of the higher exposure bundles initially inspected contained significant numbers of pins with visual standards of 4, 5, and 6. The inspections were performed by GE specialists who identified the cause of the failures as being due to CILC. The corrosion in the reload-1 fuel pins was unexpected since the Zircaloy cladding had been heat treated and belt polished to increase resistance to CILC. As a result of the initial inspection of the reload-1 fuel assemblies, PECo announced on February 15, 1989, that because of the cladding failures in some reload-1 fuel assemblies, almost all of the 268 reload-1 fuel assemblies also would be replaced with new fuel assemblies purchased for Unit 2. This is the first incident of CILC failures in heat-treated cladding. Previously, CILC attack had not occurred until the fuel exceeded at least 15,000 MWD/T exposure. Because Unit 1 was operated at reduced power to limit off-gas activity, the peak exposures were only about 10,500 MWD/T. In the reload-1 fuel assemblies that were only exposed for one cycle, CILC proceeded at a much faster rate in some fuel rods than that experienced in other plants with this type of fuel failure. In letters dated March 22 and April 3, 1989, the licensee informed the staff of the following observations from the inspection.

- In the same fuel assembly, there were rods with significant crud buildup and nodular corrosion, where adjacent rods were almost unaffected, even though the tubing was from the same heat and given the same heat treatment.

- In the reload-1 high-duty fuel assemblies (peak of 11 kW/ft), the central UO_2 pins showed the highest corrosion.

- In the reload-1 low-duty fuel assemblies (peak of 7 kW/ft), only 1 percent of the 1569 pins examined were visual standard 4 and no pins had visual standards of 5 or 6.

- In the initial core, the gadolinia rods showed the highest corrosion in the non-heat-treated population while the central UO_2 rods showed the highest corrosion in the heat treated population.

As stated earlier, before Unit 1 was shut down in January 1989, the licensee had planned to reconstitute 84 fuel assemblies discharged during the first refueling outage, replacing non-heat-treated rods with heat-treated rods. The licensee had conservatively established the criteria that only fuel meeting visual standards 1 or 2 would be reinserted for Cycle 3 restart. There were not sufficient doner pins to meet these criteria along with the requisite nuclear characteristics. The licensee revised its criteria so that non-heat-treated rods could be used, all rods would be inspected, and heat-treated rods

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*Visual standards 1 through 6 are a GE classification to define the degree of corrosion on fuel pins. Visual standards 1 and 2 represent minor oxidation and are considered acceptable for reuse. Visual standards 5 and 6 represent severe corrosion, spalling, pitting and cratering and are not considered acceptable for reuse.

that did not meet visual standards 1 or 2 would be replaced. As a result, the licensee only reconstituted 48 of the 2.48-percent-enriched initial core bundles and two reload-1 bundles. After a thorough inspection, the licensee cleared 42 of the reload-1 bundles (out of 268) for reinsertion into the core for Cycle 3. The other initial core and reload-1 bundles that were in Cycle 2 will be discharged. The remainder of the core will consist of 152 fuel bundles previously discharged from the initial core, the 224 fresh fuel bundles originally scheduled to be inserted, and an additional 296 lower enriched fresh fuel bundles that are on site and that originally were planned to have been loaded in the Unit 2 initial core.

Limerick Unit 1 was not the only BWR/4 to experience CILC in recent years. Both Hatch units had fuel failures attributed to corrosion in 1987. Hatch Unit 1 refueled in the spring of 1987 with heat treated barrier fuel and has not experienced subsequent fuel rod leaks during Cycles 11 or 12. Hatch Unit 2 refueled in the spring of 1988 with similar fuel and has operated without subsequent fuel failures during Cycle 8 (see NUREG/CR-3950). Hatch Unit 2 controlled off-gas releases by operating at reduced power. GE has identified 15 domestic and foreign BWRs, including Limerick Units 1 and 2, as being plants susceptible to CILC. According to GE, the characteristics that make these plants susceptible are:

- high-power density (greater than 49 kW/l)
- presence of a copper alloy in the condenser (Limerick Units 1 and 2 condenser tubes are Admiralty metal)
- use of filter demineralizer (vs. deep-bed demineralizers) in the condensate cleanup system

In the early Vermont Yankee, Hatch, and Browns Ferry fuel assembly failures, a majority of the failures were found in gadolinia rods. The CILC mechanism requires formation of a high density of oxide nodules. Zircaloy appears to be most susceptible to nodule nucleation in high-radiation fields at low heat flux, which is typical initial duty for (U, Gd)O_2 fuel rod cladding. Subsequent experience has shown that failures are as likely to occur in UO_2 rods as in gadolinia rods. The metallurgy of the cladding is considered to play a role in the susceptibility of the cladding to CILC, but it is not the primary factor. The cladding of 13 out of 17 failed rods in Vermont Yankee Cycle 5 were shown to be from one fabrication lot. An accelerated corrosion test of an archive sample indicated that the lot was susceptible to corrosion. Yet fuel rods from the same fabrication lot of cladding that were operated in the BWR/4 Muehleberg KKM (Switzerland), showed no corrosion attack (EPRI NP-3789).

On the basis of available data, the EPRI study concluded that the contributors to copper crud formation leading to CILC failures are:

- high copper ion content in the feedwater and reactor water (This is only likely to occur in plants using filter-demineralizers for condensate treatment vs. deep-bed demineralizers.)
- susceptibility of some clad lots to corrosion

- local thermal-hydraulic and chemistry conditions at the clad surface (The presence of other impurities in the water, such as possibly iron or organics, may assist in precipitating the unstable copper ion from solution, resulting in its accumulation on fuel surfaces. These impurities can turn loose "benign" low-thermal resistance crud into hard, dense high-thermal-resistance crud and the dense crud in turn can raise clad temperature significantly.)

- as fabricated clad surface condition

- unyielding, dense adherent, copper-crud deposits, which tend to form cracks perpendicular to the heat transfer direction, due to clad diameter changes from PCMI or thermal cycling, or both (The thermal resistance of the cracked crud in turn raises the clad temperature to a level of high oxidation rates.)

The safety significance of CILC is that it can perforate the fuel cladding, which is the first barrier to the release of fission products. However, the development of a leaking pin is readily detected by an increase in off-gas activity and the amount of off-gas activity can be controlled by reducing power, particularly the power levels in the leaking bundles. While experience has demonstrated that plants such as Limerick Units 1 and 2 can be safely operated with a modest number of fuel failures, overall safety is enhanced by preventing the formation of the copper crud that can lead to fuel failures.

5 REACTOR COOLANT SYSTEMS

5.2 <u>Integrity of the Reactor Coolant Pressure Boundary</u>

5.2.4 RCPB Inservice Inspection and Testing

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

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

In SSER 3 and SSER 5, the staff concluded that the preservice inspection program for Unit 1 was in compliance with 10 CFR 50.55a(g)(2). The staff also determined that certain preservice examinations were impractical and that relief from ASME Code requirements should be allowed for those requirements where the licensee had demonstrated that either (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with the requirements would result in hardships or unusual difficulties without a compensatory increase in the level of quality and safety. In the SER and SSER 3, the staff noted that neither the inservice inspection (ISI) program for Unit 1 or the preservice inspection program for Unit 2 had been submitted. License condition 2.C(7) required that the licensee submit the ISI program for Unit 1 by October 26, 1985.

On October 23, 1985, the licensee submitted the first 10-year interval ISI Program Plan and the Augmented Inservice Inspection Program (AISI) Plan for Unit 1. Additional information was provided by letters dated August 14, 1986, January 30, 1987 (which submitted Revision 2 to the ISI and AISI program plans), and August 11, 1987 (which submitted Revision 3 to the ISI and AISI program plans). The submittals included requests for relief from certain ASME Code requirements that the licensee had determined were impractical to perform.

The staff reviewed the ISI and AISI program plans for Limerick Unit 1 and concluded that the plans were acceptable and in compliance with the requirements of 10 CFR 50.55a(g). The staff's safety evaluation was included with a letter to the licensee dated March 4, 1988. In its letter the staff advised the licensee that it had concluded that for each of the licensee's ASME Code relief requests: (1) adequate justification had been provided for not being able to fully comply with the Code requirements, (2) the alternative examinations proposed will ensure an acceptable level of inservice structural integrity, and (3) except for the three reactor pressure vessel shell welds, all of the requested relief was granted. With regard to the three pressure vessel welds, the staff is generically reviewing the examination of pressure vessel welds with the BWR Owners Group. The licensee was advised that the requested relief for the three welds was pending and would be reviewed as part of the generic assessment.

5.2.4.4 Evaluation of Compliance With 10 CFR 50.55a(g) for Unit 2

In the SER and SSER 3, the staff addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). Since the previous supplement, the staff has completed its review of information with regard to compliance for Unit 2 submitted by the licensee in letters dated September 10, 1987; August 15, and October 27, 1988; and January 3, 1989.

Revision 2 of the preservice inspection program incorporated requests for relief from the ASME Code Section XI requirements that the licensee determined to be impractical for preservice examinations at Limerick Unit 2 and also included color-coded P&ID boundary diagrams identifying ASME Code Class 1, 2, and 3 piping systems.

Based upon the construction permit date of June 19, 1974, for Unit 2, 10 CFR 50.55a(g)(2) requires that the preservice inspection program be developed and implemented using the preservice examination requirements set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda in effect 6 months prior to the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent Code editions and addenda which are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed therein.

The licensee has prepared the Unit 2 preservice inspection program plans to meet the requirements of the 1980 Edition, Winter 1981 Addenda (80W81) of the ASME Code Section XI except that, as required by 10 CFR 50.55a(b), the extent of examination for Code Class 2 piping welds in residual heat removal, emergency core cooling, and containment heat removal systems have been determined by the 1974 Edition through Summer 1975 Addenda (74S75). The use of these later referenced Code editions is acceptable as specified by 10 CFR 50.55a(g)(3).

The "Limerick Generating Station, Unit 2, Preservice Inspection Program," through Revision 2, dated December 27, 1988, contains sections that define the scope of examinations, identify the areas to be examined, and discuss augmented inspections. This document also contains color-coded P&ID boundary diagrams in Appendix A and the requests for relief from the ASME Code Section XI requirements that the licensee determined to be impractical for systems and components within the reactor coolant pressure boundary in Appendix B.

The eight volume "Limerick Generating Station, Unit 2, Preservice Inspection Examination Plan for Nuclear Piping Systems," Revision 5, contains tables and isometric drawings that identify the specific piping system welds being examined. These tables also include a description of the item being examined, the applicable ASME Code examination categories and item numbers, the preservice examinations being performed, the calibration block required for volumetric examination (if applicable), and a notes column for special instructions.

In addition to the Code-required examinations, augmented examinations are being performed on the nuclear piping systems, as outlined by Mechanical Engineering Branch Technical Position MEB 3-1 for high-energy piping between containment isolation valves and the first outboard restraint for which no breaks are postulated. These augmented inspections provide 100-percent volumetric examination of all circumferential and longitudinal pipe welds within the "no break" boundaries and are being performed in accordance with ASME Code Section XI, with the exception that the weld and required volume is being extended to include essentially 100 percent of the through-wall thickness.

The examination plan for the reactor pressure vessel, "Limerick Generating Station, Unit 2, Reactor Pressure Vessel Preservice Inspection Examination Plan," specifies how the details of the requirements of ASME Code Section XI and the augmented requirements of RG 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," and NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," are to be met. Included are the details relating to component identification, location, procedures, calibration blocks, special equipment, and other items necessary to perform the examinations and pressure testing of the reactor pressure vessel.

Preservice ultrasonic examinations of the Unit 2 reactor pressure vessel are being performed in accordance with RG 1.150, Revision 1, Appendix A, dated February 1983. This document outlines acceptable methods for remote automatic and manual ultrasonic equipment certification and calibration, recording and sizing of indications, and reporting of examination results.

Preservice inspections of the Unit 2 reactor pressure vessel feedwater nozzles include the examinations specified in NUREG-0619, Section 4.3, for the triple-sleeve sparger design. The control rod drive (CRD) return line nozzles are not used at Limerick. The preservice inspection of the capped CRD nozzles will include a final liquid penetrant examination of the inside nozzle blend radius and bore regions and the reactor vessel wall area beneath the nozzle. Thereafter, the capped CRD nozzles will be examined by the licensee in accordance with Section XI only.

The above documents describe a preservice examination of ASME Code Class 1 components that includes essentially 100 percent of the pressure retaining welds that are not excluded from examination based on Paragraph IWB-1220 of ASME Code Section XI. On the basis of review of this information, the staff finds the sample of ASME welds selected for preservice examination within the reactor coolant pressure boundary acceptable.

Requests for relief from the ASME Code Section XI requirements, which the licensee determined to be impractical for systems and components within the reactor coolant pressure boundary, were identified in Revision 2 of the preservice inspection program. All relief requests were supported by information pursuant to 10 CFR 50.55a(a)(3). Therefore, the staff evaluated the requests for relief from the ASME Code-required examinations that the licensee determined to be impractical and concluded that the licensee has demonstrated that either (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with the specific requirements of Section XI would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

On the basis of its review of the licensee's submittals and the granting of relief from these preservice examination requirements, the staff concludes that the preservice inspection program for the reactor coolant pressure boundary at Unit 2 is acceptable and in compliance with 10 CFR 50.55a(g)(2). The detailed evaluation supporting this conclusion is provided in Appendix Q to this report.

The initial inservice inspection program plan has not been submitted by the licensee. The staff requires that this plan be submitted within 6 months from the date of issuance of the operating license. The inservice inspection program plan will be evaluated based on 10 CFR 50.55a(g)(4), which requires that the initial 120-month inspection interval comply with the requirements in the latest edition and addenda of Section XI of the Code incorporated by reference in Paragraph 50.55a(b) on the date 12 months prior to the date of issuance of the operating license. This plan will be evaluated after the applicable ASME Code edition and addenda can be determined and before the first refueling outage when inservice inspection commences.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.3 Secondary Containment Functional Design

SGTS Connection to Refueling Floor

In SSER 2 and 3, the staff approved the licensee's request for a schedular exemption with regard to connecting the standby gas treatment system (SGTS) to the refueling floor. The licensee had proposed deferring the tie-in of the SGTS to the refueling floor until the first refueling outage. At Limerick, as with most BWRs, the refueling floor is common to both units, is located above the two reactors, and is completely isolated from the secondary containment zones of Units 1 and 2 during operation. The staff concluded that this connection was not required prior to movement of irradiated fuel. A condition (2.C.14) was added to the license requiring that prior to any movement of irradiated fuel within the refueling floor volume, the licensee shall complete and test all modifications required to connect the refueling floor volume to the SGTS.

Limerick Unit 1 was shut down for the first refueling outage on May 15, 1987. Prior to movement of any irradiated fuel, the SGTS was tied into the refueling floor as verified by NRC Inspection Report 50-352/87-13 dated July 29, 1987. By letter dated January 13, 1987, the licensee proposed (1) adding prefilters in the system to extend the life of the SGTS high-efficiency particulate air (HEPA) filters, (2) adding two new SGTS fans with higher capacity (8400 cfm) to replace the existing fans (3000 cfm) to decrease drawdown time to 2.25 minutes, and (3) appropriating changes to the Technical Specifications to add new surveillance requirements and initiation and isolation systems. The modifications and changes to the Technical Specifications were approved by Amendment 6 to Operating License NPF-39 issued July 8, 1987. The satisfactory completion of the modification during the May to August 1987 first refueling outage along with appropriate flow balancing logic system functioning and other testing were verified in Inspection Report 50-352/87-18 dated September 2, 1987. These actions verified that License Condition 2.C.14 was acceptably completed and closed out the open issue in SSER 2 and 3 as being fully resolved.

6.6 Inservice Inspection of Class 2 and 3 Components

6.6.4 Evaluation of Compliance with 10 CFR 50.55a(g) for Unit 2

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

In the SER the staff addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). Since the previous supplement, the staff has completed its review of the information with regard to compliance for Unit 2 submitted by the licensee in letters dated September 10, 1987; August 15 and October 27, 1988; and January 3, 1989.

Revision 2 of the preservice inspection program incorporated requests for relief from the ASME Code Section XI requirements that the licensee has determined to be impractical for preservice examinations at Limerick Unit 2 and also included color-coded P&ID boundary diagrams identifying ASME Code Class 1, 2, and 3 piping systems.

Based upon the construction permit date of June 19, 1974, for Unit 2, 10 CFR 50.55a(g)(2) requires that the preservice inspection program be developed and implemented using the preservice examination requirements set forth in editions of Section XI of the ASME Code and addenda in effect 6 months prior to the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent Code editions and addenda which are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed therein.

The licensee has prepared the Unit 2 preservice inspection program plans to meet the requirements of the 1980 Edition, Winter 1981 Addenda (80W81) of the ASME Code Section XI except that, as required by 10 CFR 50.55a(b), the extent of examination for Code Class 2 piping welds in residual heat removal, emergency core cooling, and containment heat removal systems have been determined by the 1974 Edition through Summer 1975 Addenda (74S75). The use of these later referenced Code editions is acceptable as specified by 10 CFR 50.55a(g)(3).

The "Limerick Generating Station, Unit 2, Preservice Inspection Program," through Revision 2, dated December 27, 1988, contains sections that define the scope of examinations, identify the areas to be examined, and discuss augmented inspections. This document also contains color-coded P&ID boundary diagrams in Appendix A and the requests for relief from the ASME Code Section XI requirements that the licensee has determined to be impractical for Class 2 and 3 systems and components in Appendix B.

The eight volume "Limerick Generating Station, Unit 2, Preservice Inspection Examination Plan for Nuclear Piping Systems," Revision 5, contains tables and isometric drawings that identify the specific piping system welds being examined. These tables also include a description of the item being examined, the applicable ASME Code examination categories and item numbers, the preservice examinations being performed, the calibration block required for volumetric examination (if applicable), and a notes column for special instructions.

In addition to the Code required examinations, augmented examinations are being performed on the nuclear piping systems, as outlined by Mechanical Engineering Branch Technical Position MEB 3-1 for high-energy piping between containment isolation valves and the first outboard restraint for which no breaks are postulated. These augmented inspections provide 100-percent volumetric examination of all circumferential and longitudinal pipe welds within the "no break" boundaries and are being performed in accordance with ASME Code Section XI, with the exception that the weld and required volume are being extended to include essentially 100 percent of the through-wall thickness.

On the basis of its review of the above information, the staff finds the sample of ASME Code Class 2 welds selected for preservice examination acceptable.

Requests for relief from the ASME Code Section XI requirements, which the licensee determined to be impractical for Class 2 and Class 3 systems and components, are identified in Revision 2 of the preservice inspection program. All of the relief requests were supported by information pursuant to 10 CFR 50.55a(a)(3). Therefore, the staff evaluated the requests for relief from the ASME Code-required examinations that the licensee determined to be impractical and concluded that the licensee demonstrated that either (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with the specific requirements of Section XI would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

On the basis of its review of the licensee's submittals and the granting of relief from these preservice examination requirements, the staff concludes that the preservice inspection program for Code Class 2 and 3 systems and components at Unit 2 is acceptable and in compliance with 10 CFR 50.55a(g)(2). The detailed evaluation supporting this conclusion is provided in Appendix Q to this report.

The initial inservice inspection program plan has not been submitted by the licensee. The staff requires that this plan be submitted within 6 months from the date of issuance of the operating license. The inservice inspection program plan will be evaluated based on 10 CFR 50.55a(g)(4), which requires that the initial 120-month inspection interval comply with the requirements in the latest edition and addenda of Section XI of the Code incorporated by reference in Paragraph 50.55a(b) on the date 12 months prior to the date of issuance of the operating license. This plan will be evaluated after the applicable ASME Code edition and addenda can be determined and before the first refueling outage when inservice inspection commences.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.2 Specific Findings

7.2.2.5 Anticipated Transients Without Scram

In SSER 3, the staff noted that on June 26, 1984, the Commission published a final rule, 10 CFR 50.62, regarding the reduction of risk from anticipated transients without scram (ATWS) events for light-water cooled nuclear power plants. At the same time, the Commission directed the staff to complete and issue in the form of a generic letter explicit quality assurance (QA) guidance for non-safety-related equipment encompassed by the ATWS rule. Section 50.62(d) of the ATWS rule requires that each licensee develop and submit a proposed schedule for meeting the requirements of the rule within 180 days after issuance of the QA guidance.

Generic Letter 85-06 was issued April 16, 1985. The enclosure to the letter provided the quality assurance guidance required by 10 CFR 50.62. The generic letter specifically stated that issuance of the QA guidance in the enclosure shall be considered the reference date initiating the schedule in 10 CFR 50.62(d).

For boiling water reactors such as Limerick Units 1 and 2, the requirements of the ATWS rule were to install an alternate rod injection (ARI) system and a standby liquid control system (SLCS) and to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS.

By letter dated October 17, 1985, the licensee provided design information on the systems it had installed in Unit 1 and that it planned to install in Unit 2 to meet the ATWS rule. In response to a staff request of January 23, 1987, the licensee provided drawings and results of testing of the system by letter dated April 23, 1987.

For Units 1 and 2, the design incorporated the redundant reactivity control system in conjunction with selected equipment in the control rod drive, the reactor recirculation, and standby liquid control (SLC) systems for ATWS prevention and mitigation.

By letter dated November 3, 1987, the staff informed the licensee that it had reviewed the information provided and had concluded that the ARI design, the ATWS recirculation pump trip design, and the SLC system design comply with the requirements of 10 CFR 50.62 and the guidance published in the Federal Register on June 26, 1984 (49 FR 26036). The staff also concluded that the systems installed in Units 1 and 2 acceptably meet all of the requirements of the ATWS rule.

7.3 Engineered Safety Features Systems

7.3.1 System Descriptions

7.3.1.11 Emergency Service Water System

In the SER the staff indicated that the emergency service water (ESW) system consists of two independent loops with two 50-percent-capacity pumps in each loop. The licensee provided a clarification on December 22, 1988, which indicates that the pumps are actually 100-percent capacity for each loop, 50-percent capacity for the system. Additional information is provided in Section 9.2.1 of this supplement.

8 ELECTRIC POWER SYSTEMS

8.3 Onsite Emergency Power System

8.3.1 AC Power System

In the SER the staff indicated that each diesel generator is automatically started by either a safety injection actuation or by an emergency bus undervoltage signal on its respective bus. The applicant provided a clarification on December 22, 1988, indicating that the diesel generator automatic start is generated by either a loss-of-coolant-accident (LOCA) signal in the unit with which the diesel generator is associated or by an emergency bus undervoltage signal on its respective bus.

The staff further indicated that on the unit without a LOCA, which is also experiencing a loss of offsite power, all the Class 1E 4.16-kV breakers except the Class 1E load center feeder breakers are tripped. The licensee's clarification of December 22, 1988, indicates that on the unit without a LOCA, but with a loss of offsite power, the 4.16-kV breakers for the common loads that are required to support the unit with a LOCA are not tripped and the common loads are automatically restarted as required.

The staff further discussed the emergency override of the test mode and bypassing of the protective trips. The licensee's clarification of December 22, 1988, indicates that the override and trip bypasses are applicable only to the unit experiencing a LOCA.

The staff reviewed these clarifications and finds the application of the criteria discussed in the original SER to the unit experiencing a LOCA as appropriate and consistent with the regulations, standards, and guidance, and therefore acceptable.

8.3.2 DC Power System

In the SER the staff states that each Class 1E dc battery is separately housed in a ventilated room apart from its charger, motor control center, and distribution panel. This is not correct. During a plant walkdown of Limerick Unit 1 that compared the staff's SER with the as-built configuration and the FSAR, the resident inspectors noted that the actual battery room configuration has the battery chargers, the fuse panels (with switched disconnects), and in one case a dc distribution panel located within the battery rooms. Section 8.3.2.1.1.5 of the FSAR states: "For each Class 1E dc system, the battery bank, chargers, and dc switchgear are located in separate compartments of the seismic Category I control structure. The battery compartments are ventilated by a system that is designed to preclude the possibility of hydrogen accumulation." The licensee has stated that the FSAR description intention is that the four battery divisions are in separate compartments, not that the components within a battery division are in separate rooms (see Inspection Report 50-352/88-20 dated November 30, 1988).

Although the plant configuration is not in conformance with the Institute of Electrical and Electronics Engineers (IEEE) Standard 484-1975, the staff concludes the arrangement is acceptable provided administrative procedures ensure that there is adequate ventilation to limit hydrogen accumulation, that there are periodic functional tests of the ventilation system, and that appropriate actions are taken in the event that the ventilation system becomes inoperable.

8.4.2 Nonsafety Loads on Emergency Sources

In the SER the staff indicated that isolation devices are provided to trip non-Class 1E loads from the Class 1E buses upon detection of a LOCA signal. The licensee provided clarification on December 22, 1988, that these trips occur only on the unit experiencing the LOCA. The staff has reviewed this clarification and finds this application of automatic disconnection of nonsafety loads connected to the Class 1E buses upon detection of a LOCA signal to be in accordance with RG 1.75 and therefore acceptable.

9 AUXILIARY SYSTEMS

9.2 Water Systems

9.2.1 Emergency Service Water System

In the SER the staff described the emergency service water (ESW) system as consisting of two piping loops for the two-unit plant and containing two 50-percent-capacity pumps per loop. It also discussed the planned two-unit operational scheme as Unit 1 pumps being powered from the A and C diesels and the Unit 2 pumps being powered from the B and D diesels associated with their respective units. Finally, it indicated that each ESW pump could be powered by either one of two diesels (e.g., diesels A from Unit 1 or Unit 2). The licensee provided clarification on each of these points in its submittal of December 22, 1988, as indicated below.

The ESW system consists of two piping loops for the two-unit plant. Each loop has two 100-percent-capacity pumps per loop. One pump in each loop is powered by Unit 1 and one pump in each loop is powered by Unit 2. The A and B ESW pumps are powered by Unit 1 diesel generators A and B, and the C and D ESW pumps are powered by Unit 2 diesel generators C and D. The staff has reviewed these clarifications and finds the design to continue to meet the applicable requirements and guidance of SRP 9.2.1 and therefore acceptable.

9.2.8 Control Structure Chilled Water System

In the SER the staff described the control structure chilled water (CSCW) system and implied, that after Unit 2 was licensed, one of the two 100-percent-capacity systems would be powered from Unit 2. On December 22, 1988, the licensee provided a clarification that both CSCW systems will be powered from Unit 1 even after Unit 2 is licensed. The staff has reviewed this design in conjunction with the control structure ventilation system and, because the Limerick control room is shared by both units, finds the design in compliance with SRP 9.2.1 and therefore acceptable.

9.4 Heating, Ventilation, and Air Conditioning Systems

9.4.1 Control Structure Ventilation System

In the SER the staff described the control structure ventilation (CSV) systems as shared by both units with one train powered by one unit and the redundant train powered by the other unit. The licensee provided a clarification of the two-unit power supply scheme in its submittal of December 22, 1988. Both CSV systems will be powered from Unit 1 even after Unit 2 is licensed. The staff has reviewed this design in conjunction with the control structure chilled water system and, because the Limerick control room is shared by both units, finds the design in compliance with SRP 9.4.1 and therefore acceptable.

9.4.5 Engineered Safety Feature Ventilation System

Spray Pond Pump Structure Ventilation System (FSAR Section 9.4.7)

The staff discussed the spray pond pump structure and its ventilation system in the SER. The requirements of GDC 5 were not considered applicable. The licensee provided clarification in its submittal of December 22, 1988. The staff has again reviewed the system and identified the following corrections. The heaters in the spray pond pump structure ventilation system are normally powered by a Class 1E source but are automatically tripped upon receipt of a LOCA signal in the unit from which they are powered. Also, the RHR service water pumps are manually started as discussed in Section 7.3.1.12 of the SER. Further, because the ESW and RHR service water pumps are shared between the units, the spray pond pump structure ventilation system also is considered as a shared system. The two ventilation systems are completely independent, and if one system fails to perform its function, the standby system is automatically started. Therefore, the requirements of GDC 5 are satisfied. The staff considers these corrections to have no effect on the conclusions of the SER and the system is therefore acceptable.

10 STEAM AND POWER-CONVERSION SYSTEM

10.4 Other Features

10.4.6 Condensate Filter Demineralizer System

In the SER the staff stated that Limerick Units 1 and 2 each have eight Graver "Powdex" filter demineralizers to polish 100 percent of the condensate flow. Seven are normally in service and one is on standby for replacing an inservice unit at the end of its useful life.

Powdex is a precoat process that employs a medium of powdered cation and anion ion exchange resin either neat or in combination with fibrous material. The precoat medium is mixed in water to form a pumpable slurry and applied to tubular filter septa. Untreated condensate, which flows through the precoat layer during service, is simultaneously filtered and demineralized by this process. Compared to deep-bed demineralizers, the Powdex units have a limited ion exchange capacity for soluble copper and iron ions.

As discussed in Section 4.2.3.1 of this supplement and in the licensee's submittal of April 3, 1989, there are a number of factors that affect the propensity for crud-induced localized corrosion (CILC) of Zircaloy fuel cladding. These include the tubing material manufacturing process, the thermal/mechanical/nuclear duty, and the water chemistry environment. When the extent of failures in the fuel assemblies from the first reload of Limerick Unit 1 were discovered in February 1989, PECo and GE formed joint study teams to investigate the root cause, particularly since this was the first time CILC failures were encountered in heat-treated cladding. The PECo/GE teams concluded that none of the thermal/mechanical/nuclear design differences between the reload-1 fuel and the initial fuel were likely to have contributed directly to the corrosion performance. There are also design constraints on these factors if the fuel is to serve its intended function of producing power. The teams also concluded from the study of domestic and foreign BWRs that the CILC failures in Unit 1 proceeded at a much faster rate than expected or previously encountered and that the cause was not exclusively as a result of the cladding metallurgy and not exclusively as a result of the high copper levels that had existed through most of Cycle 1. The teams did not pinpoint what might be the threshold levels on the factors that are suspected of causing CILC.

GE has identified 15 domestic and foreign plants as having the characteristics of being CILC susceptible. These characteristics are:

- high-power density (greater than 49 kW/l)
- presence of copper alloy in condenser
- Filter demineralizer (vs. deep-bed demineralizer) for condensate cleanup

The power density cannot be significantly changed without affecting power output. Replacing the condensers with titanium tubed units is an expensive, major modification, although some BWRs such as Brunswick Units 1 and 2 have titanium tubed condensers and other BWR facilities such as Hatch have announced their intention to replace the condensers. Of the three characteristics, the least expensive and simplest to change is to replace the filter demineralizers with full-flow deep-bed demineralizers, provided sufficient space is available to house the large units.

For CILC to occur, copper has to be present in the coolant. The threshold level to preclude deposit buildup on the fuel cladding is not well defined and varies with coolant chemistry (particularly pH); the presence of other contaminants such as iron, oil, resins, and other organics; local thermal hydraulic conditions at the cladding surface; and the as-fabricated cladding surface condition.

Iron is suspected as having a synergistic effect with copper in promoting formation of unyielding, dense, adherent deposits on fuel elements. From tests at Chalk River in the early days of the Naval Reactors Program, it has been known that corrosion products in the coolant deposited on the fuel elements. Even in BWRs without copper, high concentrations of iron in the coolant can result in significant crud deposits on the fuel elements. Brunswick Units 1 and 2 have titanium tubed condensers with carbon steel condensate and feedwater piping. The plants utilize the original filter demineralizers and deep-bed demineralizers for condensate purification. However, the plant design incorporates forward-pumped heater drains in which condensate from the three highest temperature feedwater heaters, moisture separators, and reheaters is drained to a deaerator tank and then pumped to the feedwater pump suction, bypassing the filters and demineralizers. The heater drain water makes up 30 percent of the final feedwater volume. There are a few other foreign BWRs that use this design; but, as of 1985, Brunswick Units 1 and 2 were the only operating forward-pumped BWRs in the United States. This feedwater system is similar to the secondary feedwater system of many pressurized-water reactors (PWRs). (B. C. White and J. W. Davis, "Feedwater Iron Reduction at Brunswick," 49th International Water Conference paper IWC-88-54, October 1988.)

Although forward pumping increases efficiency, it also creates some problems for plant chemistry. The corrosion products from piping, feedwater heaters, and heater drain pumps are transported directly to the reactor without the benefit of demineralization, resulting in a typical iron loading rate of 23 g/hr (1.21 lb/day). Like Limerick Units 1 and 2, Brunswick has BWR Standard Technical Specifications, which are "extremely" permissive when compared to chemistry guidelines from the Electric Power Research Institute (EPRI) and the Institute of Nuclear Power Operations (INPO). The guideline specified by EPRI for feedwater insoluble iron concentration is 5 ppb. Until the extraction steam piping was replaced during the 1987 and 1988 refueling outages, Brunswick Units 1 and 2 operated with iron concentrations at or above the EPRI guideline. Fuel deposit analysis performed at the end of the first fuel cycle for Brunswick Unit 2 showed deposit levels of 4 to 7 mg/cm^2 at the center of the core with a peripheral bundle averaging 11 mg/cm^2 with a maximum value of 36 mg/cm^2. Although these crud levels are elevated (>5 mg/cm^2 is considered high), they are not unusual for plants with forward-pumped heater drains (IWC-88-54). However, the extensive crud deposits at Brunswick did not result in fuel failures. The point is that even in the absence of copper, iron can cause deposits on fuel cladding. If copper is present, it is desirable to have condensate treatment equipment that will reduce both the iron and copper to very low levels because of the potential synergistic effect.

Limerick Unit 1 operated with relatively high levels of iron as well as copper in the feedwater during Cycle 1. Neither Limerick unit has inline analyzers for measuring iron and copper. The weekly analyses are composite samples that provide average values, but may miss transient breakthroughs. The combination of the iron and copper together may have contributed to the unexpected and accelerated buildup of deposits on the fuel elements. Iron, copper, and zinc could be more effectively removed by full-flow deep-bed demineralizers than with the present cleanup system.

By letter dated April 3, 1989, the licensee submitted a report documenting the results of its evaluation of the Unit 1 fuel failures that occurred during Cycle 2 operation. In that letter the licensee stated:

> The fuel failures were due to crud induced localized corrosion (CILC). The root cause of the failures is believed to be due to the synergistic effect of less than expected corrosion resistance of the cladding and early cycle chemical transients in conjunction with increasing levels of copper input to the reactor water. The attached report includes the corrective actions and preventive measures that are being implemented to address each of these three causal factors and to thereby avoid further CILC related fuel failures at both Units 1 and 2.

During Cycle 2, there were two significant chemical transients: (1) On September 19, 1987, shortly after startup of Cycle 2, the electrohydraulic oil piping developed a leak resulting from vibration. The oil drained into the radwaste system and was processed. The practice at Limerick is to return almost all processed water to the condensate storage tank to minimize radioactive discharges. Therefore, total organic carbon (TOC) concentrations in the feedwater were high for an extended period of time (NRC Inspection Report 50-352/87-21 dated October 22, 1987). (2) On December 24, 1987, approximately one pound of condensate filter demineralizer resin was inadvertently injected into the reactor vessel. Reactor coolant conductivity increased to a maximum of 3.62 micromhos per centimeter and pH dropped to 5.42. Both parameters were restored to technical specification limits within 6 hours (NRC Inspection Report 50-352/87-31 dated March 8, 1988).

Another potential source of TOC in the Limerick feedwater is the routine use of "Solution A," a polyacrylic acid used to produce small floc size.

Under radiation, carbonaceous material breaks down and can deposit on fuel element surfaces. Tests in a Chalk River test loop in the mid-1950's showed that the decomposition residue from ion exchange resin or oil could form an adherent coating on simulated fuel elements, resembling coke or creosote in appearance. In one military reactor, resin degradation products caused deposits on both the fuel and control rod drives to the extent that control rod movement was restricted. In these cases, the carbon deposited rather uniformly on all surfaces and showed up in the analyses of scrappings taken to investigate what solvents might be used to remove the deposits.

With the amount of oil and other organic materials used in a nuclear power plant, there is always the potential for some to get in the radwaste collection tanks. The use of TOC analyzers to monitor process streams can minimize potential contamination of the condensate. If oil does get in the condensate, the use of deep-bed demineralizers are more effective in removing TOC than filter demineralizers.

In the letter of April 3, 1989, and in a meeting held March 15, 1989, the licensee made a number of commitments regarding water chemistry and corrosion control for Limerick Units 1 and 2. These include:

- employ continuous in-line analyzers to monitor water chemistry by
  - using ion chromatographs to monitor feedwater and reactor water ionic concentrations
  - using a gas chromatograph to measure oxygen and nitrogen concentrations in the reactor water
  - using a sensitive TOC analyzer to measure TOC concentrations in the feedwater
- limit chemical transients by
  - routinely inspecting to limit oil entering radwaste collection tanks
  - limiting TOC in radwaste sample tank to less than 200 ppb
  - limiting condenser air inleakage to less than 30 scfm
  - controllling feedwater oxygen to between 20 to 50 ppb
  - evaluating procedures and training operators in filter/demineralizer backwash, slurry preparation, and precoating and testing
- minimize copper in feedwater by
  - changing filter/demineralizer elements in Unit 1
  - continuing tests with alternate resin mixes in Unit 1
  - installing full-flow, deep-bed demineralizers no later than the next refueling outages for Units 1 and 2
  - limiting copper concentration in feedwater to less than 0.2 ppb

The staff endorses the licensee's action plan to prevent future CILC failures. The licensee has put forth a comprehensive effort to evaluate the factors that may have caused the fuel failures in Limerick Unit 1. These investigations have contributed significantly to the body of knowledge on the CILC phenomena in BWRs. On the basis of the information about CILC today, the staff concludes that, in combination, the actions being taken by the fuel manufacturer to improve fuel cladding metallurgy and the actions being taken by the licensee to control water chemistry will together prevent future CILC fuel assembly failures at Limerick Units 1 and 2.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Operations

13.1.1 Organizational Structure

Significant organizational changes have occurred at both the corporate and station level since SSER 3 was issued. The licensee announced a major corporate reorganization of the company's Nuclear Operations and Support Services, which became effective on November 1, 1987. Changes included creation of a Vice President for the Limerick Station, assumed by the former Unit 1 Plant Manager; corporate vice presidents for newly formed Divisions of Nuclear Services and Nuclear Engineering; and a Senior Vice President, Nuclear. The reorganization also involved reconstituting the Nuclear Review Board (NRB) to include membership of three senior executives outside the PECo organization.

On February 2, 1988, the President and Chief Operating Officer announced his retirement effective March 1, 1988. On February 16, 1988, a new Executive Director Nuclear from outside the PECo organization was announced who subsequently became a PECo employee and assumed the title of Executive Vice President Nuclear on March 13, 1988. The company's Chairman and Chief Executive Officer announced his retirement effective April 13, the date of the licensee's annual board meeting. The Board of Directors elected a new Chairman and Chief Executive Officer, a former PECo Vice President who had recently been elected President and Chief Operating Officer at another utility.

The licensee has revised its corporate organizational structure to provide an organization dedicated only to nuclear power activities (the nuclear organization) with direct management authority and responsibility over all aspects of nuclear operations, engineering, maintenance, and construction. The new nuclear organization will be headed by an Executive Vice President-Nuclear with nuclear responsibilities only. This nuclear organization has been formed by separating nuclear engineering, maintenance, and other support activities for nuclear operations from corresponding support activities for fossil and hydro production and by reassigning the resources for these activities to the new nuclear organization. The former positions of Senior Vice President-Nuclear Power, Nuclear Production Manager, Superintendent-Nuclear Generation Division, Superintendent-Nuclear Services, and Manager-Nuclear Plant have been abolished and the functions under these positions have been reassigned within the new organization under the Executive Vice President-Nuclear.

There are three staff organizations and five line organizations that report to the Executive Vice President-Nuclear, as shown in Figure 13.1. The five line organizations have responsibility for the Limerick Generating Station, Unit 2 construction, the corporate Nuclear Engineering and Nuclear Services Division, and the operating site groups at Peach Bottom and Limerick. The three licensee staff organizations have responsibility for the corporate Nuclear Review Board (NRB), Nuclear Quality Assurance (NQA), and Organization and Management Development.

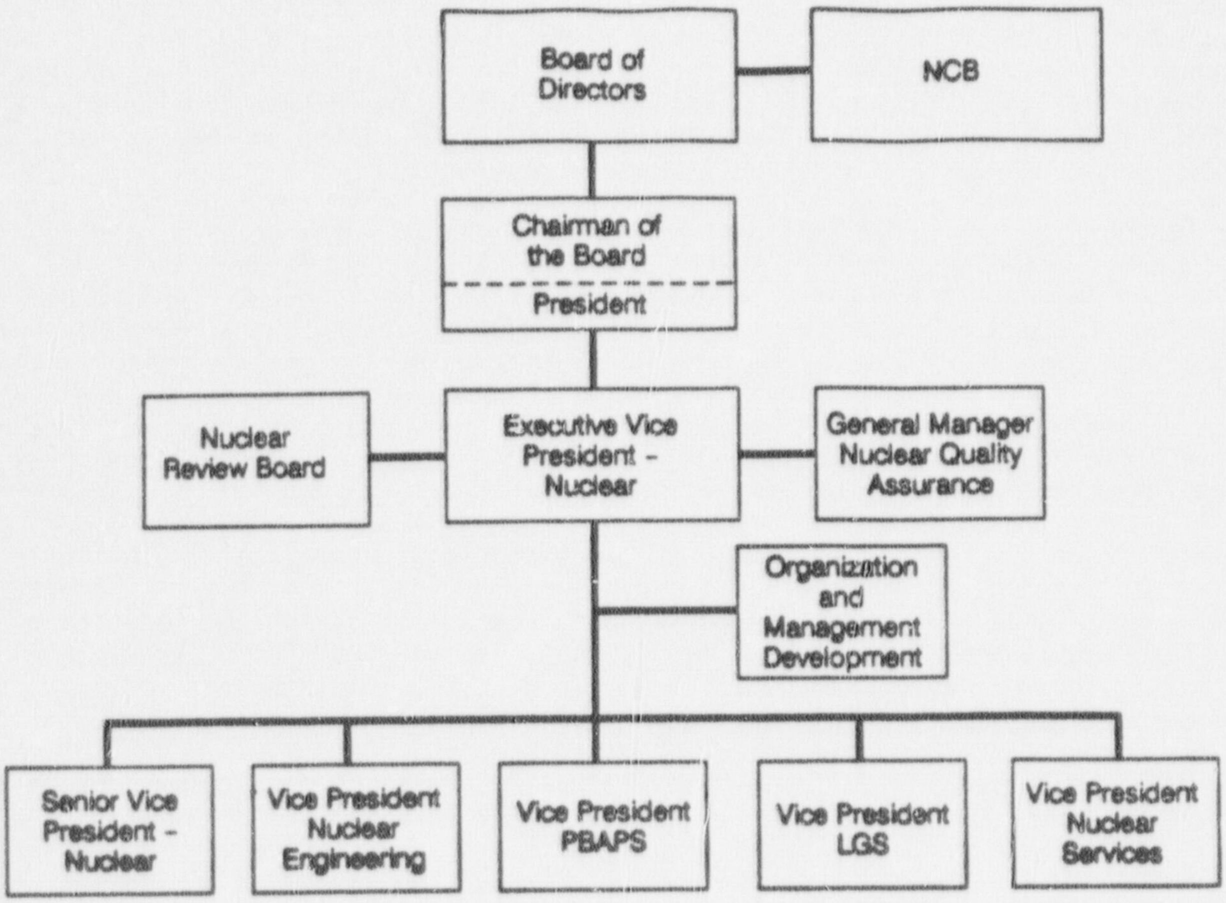

Figure 13.1 Corporate reporting structure for the nuclear organization

To strengthen the independent assessment process, the licensee has revised the NRB charter and reporting relationship and has established a Nuclear Committee of the Board of Directors. The NRB consists of nine members, six PECo personnel and three outside consultants. The NRB is a group of individuals independent of plant operations charged with providing an independent review of safety-related activities at both Peach Bottom Units 2 and 3 and Limerick Units 1 and 2. The NRB reports directly to the Executive Vice President-Nuclear and submits copies of reports to the Chief Executive Officer of PECo. In addition, the Chairman of the NRB will meet directly with the Chairman of the Nuclear Committee of the Board and will report to this board at least annually.

The staff reviewed the provisions for the NRB and finds that they meet the guidance for independent review as described in Section 13.4 of the Standard Review Plan (NUREG-0800) and are acceptable.

By letter dated October 31, 1988, the staff issued Amendment No. 10 to the Limerick Unit 1 operating license approving the revised corporate organizational structure.

13.1.2 Operating Organization

As stated above, organization changes have occurred at the station level as well as in the corporate organization since SSER 3 was issued. As part of the organizational restructuring designed to provide increased control and corporate direction on site, a corporate management position of Vice President was added at both Peach Bottom and Limerick. The Vice President-Limerick will have full authority with regard to the work of all operational organizations and all regular site operations employees except for those involved in construction of Limerick Unit 2 or in independent assessment and oversight activities. Figure 13.2 shows the revised station organizational chart. This onsite organization structure was approved by Amendment No. 10 (issued October 31, 1988) to the Limerick Unit 1 operating license.

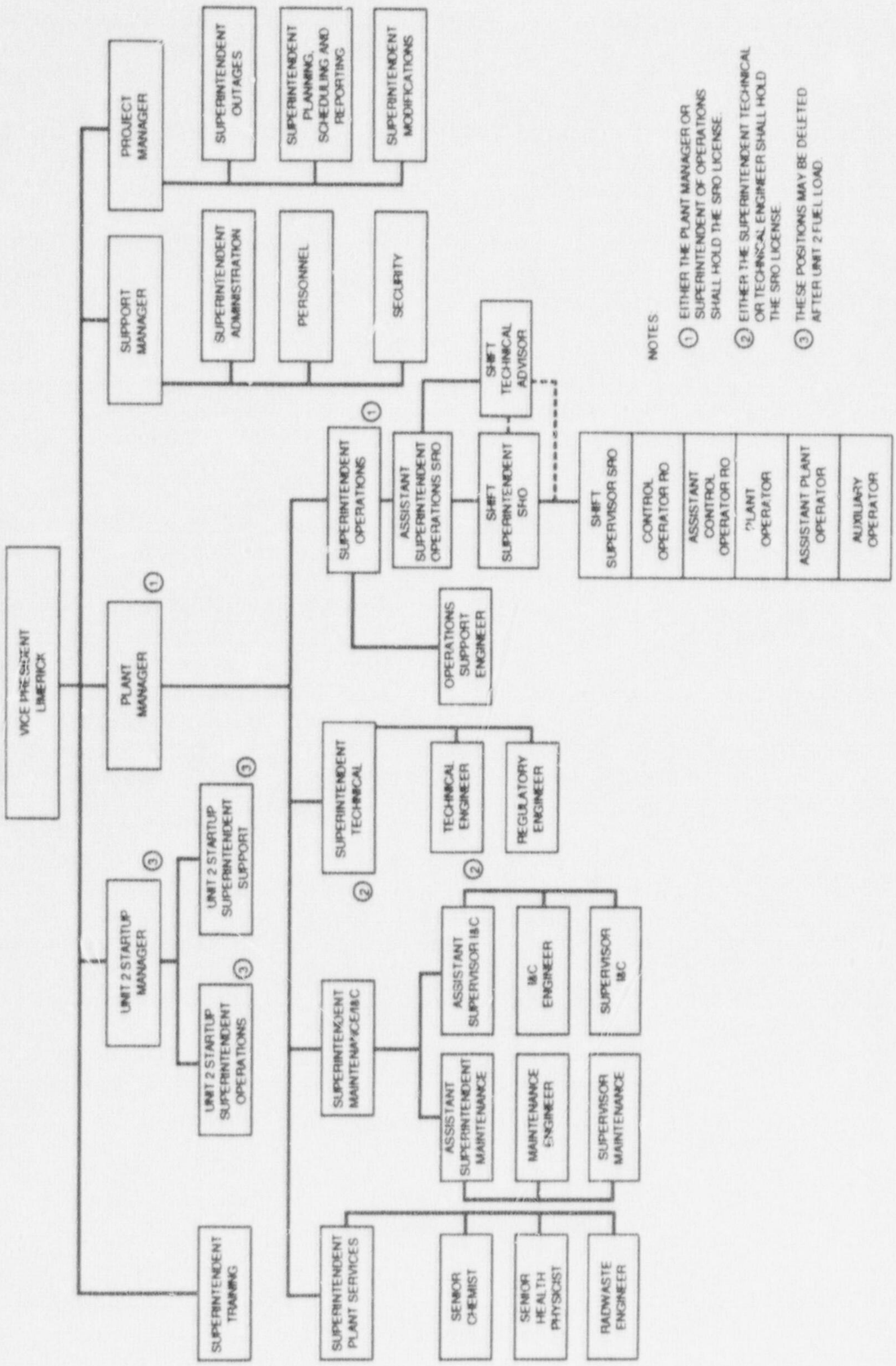

Figure 13.2 Limerick onsite organization

17 QUALITY ASSURANCE

17.1 General

The SER discusses the staff's review and evaluation of the licensee's quality assurance (QA) program for the operations phase of Limerick Generating Station Units 1 and 2 as described in Chapter 17 of the FSAR. The licensee has since revised its organization and program for QA, and the staff has reviewed these changes to determine whether the QA program description continues to be acceptable. The licensee responded to the staff's request for additional information regarding the structures, systems, and components that are under the control of the QA program. This supplement to the SER reflects the staff's review of the latest QA program description for Limerick as described in the FSAR through Revision 57.

17.2 Organization

The structure of the organization responsible for the operation of Limerick and for the establishment and execution of the operations-phase QA program is shown in Figure 17.1. FSAR Chapter 13 discusses the operations organization for Limerick.

The President of PECo has ultimate responsibility for the Limerick QA program. The President of PECo has delegated to the Executive Vice President of the Nuclear Group the responsibilities for establishing and maintaining the QA program. The Nuclear Group, under its QA program controls, obtains procurement, personnel, maintenance, and other support services from contractors or from other PECo organizations, as required.

PECo's Nuclear Quality Assurance (NQA) organization is responsible for administering the overall NQA program of the Nuclear Group. The NQA organization is headed by a General Manager who reports directly to the Executive Vice President of the Nuclear Group. The NQA organization includes approximately 85 QA engineers and auditors servicing the Limerick and Peach Bottom facilities. The major responsibilities of the NQA organization are maintenance of PECo's quality policy, administration and coordination of and overall direction for implementing the NQA program, performing audits and inspections, identifying problems adverse to quality, verifying implementation of corrective actions, and maintaining liaison with the plant staff, contractors, and other PECo organizations to provide overall direction to QA activities and problems.

The Limerick organization, under a Vice President who also reports to the Executive Vice President of the Nuclear Group, has overall responsibility for executing the administrative controls at the plant and implementing responsibilities assigned to plant personnel by the QA program. An overview of the QA program is provided by NQA personnel who are organizationally independent of the other organizations in the Nuclear Group. The NQA organization is responsible to ensure, through its auditing and quality control functions, that activities affecting quality are accomplished in accordance with the requirements of the QA program.

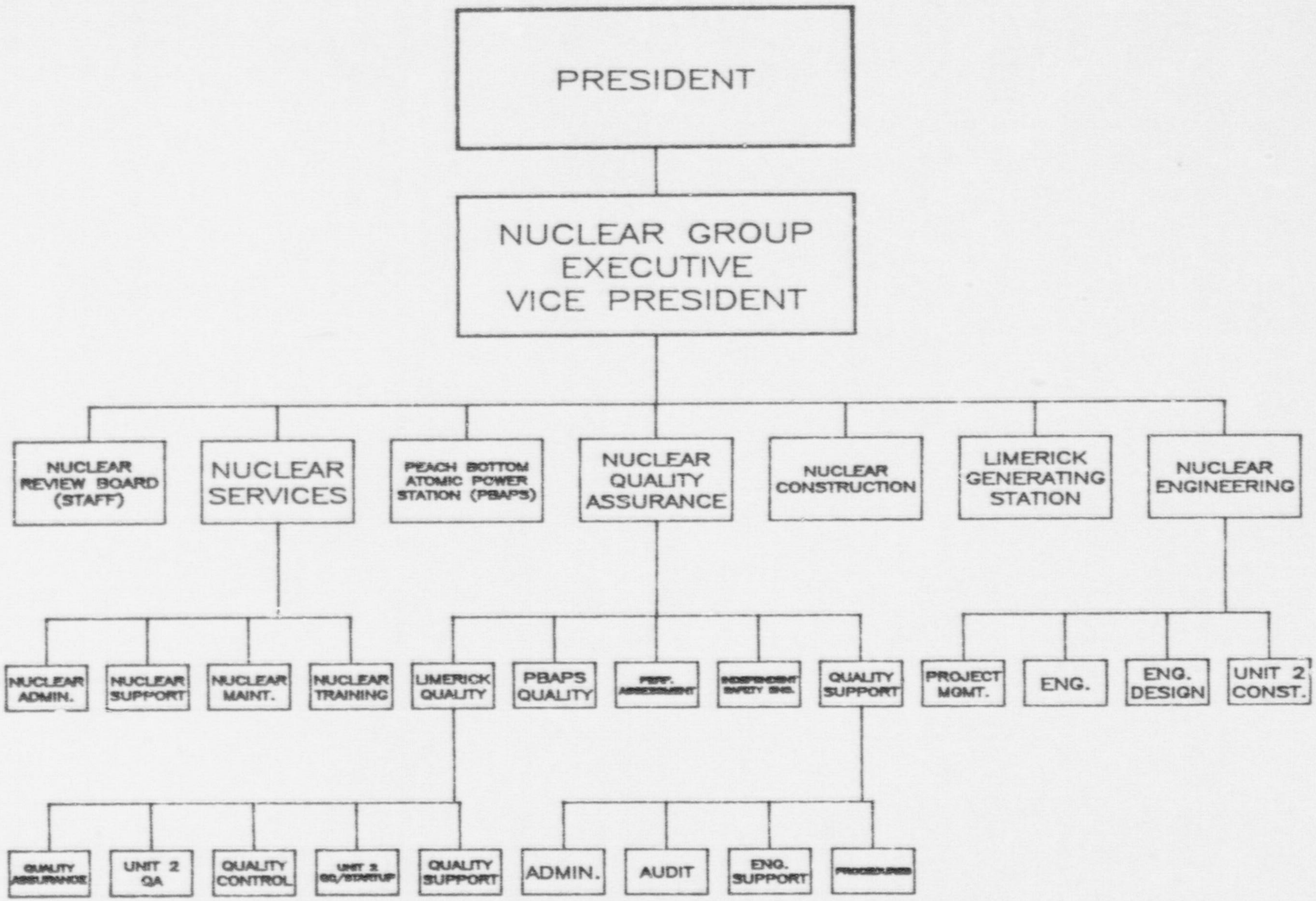

Figure 17.1 PECo organization for Limerick quality assurance

17.3 Quality Assurance Program

In the SER the staff stated that the licensee was preparing a response to the staff's request for additional information regarding the scope of the QA program. Subsequently, in FSAR Revision 32, PECo provided additional information regarding the structures, systems, and components under the control of the QA program. The staff reviewed this information and concluded that the scope of the QA program is acceptably described in the FSAR.

17.4 Conclusion

On the basis of its detailed evaluation of the QA program description in FSAR Chapter 17 (through Revision 57), the staff concludes that the organizations and persons performing QA functions have the required independence and authority to effectively carry out the QA program without undue influence from those directly responsible for cost and schedule.

The staff is pursuing resolution of 17 questions regarding the detailed implementation of the QA program. The staff will discuss the resolution of these issues in a future supplement as well as compliance with Appendix B to 10 CFR Part 50 and with the acceptance criteria in Section 17.2 of the Standard Review Plan (NUREG-0800).

17.6 Readiness Verification Program

For Limerick Unit 1, the licensee contracted for an independent design verification program (IDVP) to provide additional assurance that Unit 1 had been designed and constructed in accordance with the regulations and FSAR commitments. The results were discussed in SSERs 3 and 4.

To ensure readiness of Limerick Unit 2 for fuel loading, the licensee developed two self-assessment activities. The first activity was a readiness program assessment (RPA) to identify and assess the adequacy of existing licensee programs and processes that are intended to ensure and demonstrate completion and readiness of Unit 2 for operation in accordance with licensing commitments. The RPA was conducted in the summer of 1987 by a team of PECo and International Energy Associates Limited personnel. The results of the RPA were presented in a Readiness Program Assessment Report dated March 8, 1988, and transmitted by the licensee's letter of April 11, 1988.

The second and much more extensive activity was a readiness verification program (RVP) that provided for a comprehensive integrated process to assess the design, construction, and operational aspects of Unit 2. A major feature of the RVP associated with the design and construction review was an independent design and construction assessment (IDCA). The other major feature was an operational readiness assessment. A comprehensive description of the as-implemented RVP is contained in the report transmitted to the NRC by the licensee's letter of September 16, 1988.

The IDCA is a comprehensive technical assessment of the design and construction process implemented for Unit 2. By focusing on a system that represents a broad sample of design and construction activities, the IDCA is the equivalent of a combined NRC integrated design inspection (IDI) and construction assessment team (CAT) inspection. Specifically, it uses NRC's deep-vertical-slice techniques to perform an in-depth technical review of all design and construction activities associated with the sample system, starting with basic licensing commitments and ending with the as-installed configuration of the system and supporting structures. The depth of the assessment and the breadth of activities reviewed provides a measure of the overall design and construction process as implemented for Unit 2. Hence, the results and conclusions of the review provide both an assessment of the status of the design and construction of the sample system regarding licensing commitments and an assessment of the effectiveness of implementation of the plant design and construction process.

The proposed IDCA was outlined in the licensee's letter of May 6, 1988, and discussed in depth with the staff during several meetings. PECo contracted with Stone and the Webster Engineering Company (SWEC) of Cherry Hill, New Jersey to conduct the independent assessment. SWEC assembled two experienced teams to perform the IDCA: one to conduct the independent design assessment (IDA) and the other to conduct the independent construction assessment (ICA).

A detailed description of the proposed IDCA was provided by the licensee's letter of June 1, 1988, including the overall approach, the assessment teams, and the proposed implementation of the assessment. As a result of discussions with the staff, it was decided after evaluating a number of major systems that the objective of the IDCA, to review the adequacy of the design and construction process for Unit 2, would be best achieved by conducting an independent review of selected plant systems, components, structures, and interfaces associated with the containment heat removal (CHR) mode of operation of the residual heat removal (RHR) system. The IDCA performed an in-depth review of the RHR function of injecting water into the reactor vessel following the initiation of a postulated LOCA condition and returning water to the RHR pump from the suppression pool. The selected system operations also included subsequent cooling of the RHR heat exchanger by the RHR service water system with heat rejection to the cooling tower or the spray pond. This ensured that a system common to both units was reviewed during this program. In addition, those systems directly supporting the CHR mode were also reviewed.

The selection of the CHR mode as a combination of systems acting together in a specific safety mode provides the basis for an in-depth review of design and construction activities while focusing the IDCA teams' activities on a reasonable amount of plant equipment and structures. These systems, functioning during the CHR mode, provided the basis for the in-depth assessment of the architect engineer (A/E) design process since the following activities were accomplished within the A/E scope of services:

- interface with the NSSS design process, including postulated accident conditions
- system design for a common system, i.e., serves both units
- service water pump specification and procurement
- pipe sizing, layout, and procurement
- pipe stress analysis and support design, including ASME III, Class 1, 2, and 3 piping

- electrical power to pumps and valves and procurement of motor-operated valves (MOV) and pump motors, including power distribution equipment/components

- control logic interface with NSSS equipment

- control system logic for operation of A/E procured valves and pumps

- routing of power and control cables, including electrical separation criteria

- structural elements and support system components, including walls and floors

- amplified response spectra development and application

- hazards program design and analysis

- environmental qualification of equipment, including seismic requirements

- supporting systems for proper operation of the CHR mode, including compartment cooling and pump/motor cooling systems

- design interface with subcontractors and component suppliers

Following a meeting with the staff, primarily to discuss the protocol that would be followed to provide a really independent assessment while the IDCA was being conducted, the licensee submitted Revision 1 to the proposed IDCA by letter dated July 7, 1988. The revised program was approved by the staff's letter of July 28, 1988. As stated in the letter, the scope of the IDCA program is viewed by the staff to be a representative review of the A/E's design that will include all major design disciplines as well the design interfaces be-tween the A/E, the nuclear steam supply system vendor, and other major vendors. In a similar vein, the construction program in the IDCA will review a representative sample of all major construction attributes including component installation as well as system and component testing. Therefore, the IDCA, if properly implemented, will allow the staff to draw overall conclusions regarding the adequacy of the design and construction of Limerick Unit 2.

The NRC planned to monitor each of the design and construction aspects of the IDCA in three phases: (1) preparation of review plans, (2) implementation of the review plans and performance of the review, and (3) evaluations of the final IDCA report including assessment of the corrective actions.

During the week of August 8, 1988, a 14-member team from the NRC conducted an inspection of the proposed IDA and ICA review plans at the offices of the independent contractor, Stone and Webster Engineering Company, Cherry Hill, New Jersey. The results of the inspection were documented in Inspection Report 50-353/88-200, which was transmitted to the licensee by the staff's letter of September 29, 1988.

The NRC inspection team found the IDA review plans to be comprehensive, explicit, and logically structured and the SWEC reviewers to be experienced and technically competent. The NRC design inspection team recommended addition of certain design attributes and clarifications. The NRC construction inspection team concluded that the ICA review plans required additions and clarifications, that the scope and depth were not as well defined as for the IDA, and that training for SWEC team members had not been completed. Many of the additions and clarifications recommended by the NRC inspection team were added to the review plans by SWEC before the conduct of the exit meeting by the inspection team. All of the recommended additions and clarifications were subsequently included by PECo and SWEC. The final SWEC review plans were found to be fully acceptable.

The performance of the ICA by SWEC was assessed by two NRC team inspections. The inspections were conducted on site during August 22 through 26, and September 18 through 30, 1988. The results are documented in Inspection Report 50-353/88-202 issued January 23, 1989.

The NRC inspection team reviewed the preliminary field matrices and review plan attribute checklist generated by SWEC, reinspected a sample of items and installations reviewed by SWEC, and inspected an independent sample of construction attributes for systems and components that were outside of the scope of the IDCA program. SWEC's independent reviews of construction were generally comprehensive and met the requirements of the overall IDCA program. Their observations were properly recorded as action items or observation reports as the program required. Since the NRC inspection followed closely after the SWEC ICA, some of the SWEC audit documents had not been completed. The NRC team reinspected a sample of construction reviewed by SWEC to assess the adequacy of the ICA review. In general, the inspection team's findings confirmed SWEC's observations. The team did identify a few discrepancies not identified by SWEC; however, these were relatively minor discrepancies.

The scope of the NRC ICA inspection was broader than the scope of PECo's IDCA in that the NRC team reviewed systems and components that were outside the ICA scope. Because of this difference, the NRC team identified some problem areas that were not apparent from SWEC's ICA effort. These included damaged wire in four resistance temperature detectors, improper mounting of a pressure transmitter, and mounting bolt discrepancies on mechanical equipment. None of the SWEC or NRC observations disclosed any significant problems in quality of construction or indications of programmatic weaknesses.

By letter dated February 10, 1989, the licensee submitted the construction assessment report for Unit 2 documenting the results of the ICA portion of the IDCA performed by SWEC. Over 23,000 separate checks of hardware and documentation were made to complete this assessment. The assessment concludes "that the construction of safety-related systems and structures at Limerick 2 is satisfactory and generally in accordance with drawings and specifications." Although minor deficiencies were identified, they were determined to have no impact on the ability of components to perform their intended safety function; appropriate action was initiated to correct these deficiencies. An NRC team inspected the results presented in this report in an onsite inspection the week of March 27, 1989.

The second phase of NRC's team inspection of the IDA (i.e., review of SWEC's implementation of the design assessment) took place at SWEC's offices in Cherry Hill, New Jersey, during the week of October 3, 1988. The results of this team inspection are documented in Inspection Report 50-353/88-201 issued November 29,

1988. This inspection assessed the effectiveness of SWEC's review of design-basis documentation for Limerick Unit 2 and reviewed the completeness of the IDA review plan checklists.

The NRC inspection team observed many instances that illustrated that SWEC was identifying and aggressively pursuing the resolution of technical issues through action items and observation reports. In general, the inspection team found that SWEC was conducting the IDA in sufficient technical depth to make valid conclusions about the design of Unit 2. Also, the checklists that SWEC was completing to document the review were being filled out in sufficient detail to provide an auditable documentation trail of the IDA. However, during its review of some of the same documents reviewed by SWEC, the inspection team did identify some issues that SWEC had overlooked. SWEC agreed to include these additional items in its scope of review and in some instances completed their review while the inspection team was still on site. The inspection team also found a few instances where SWEC did not adequately review design documentation. While there were a few relatively minor deficiencies, the NRC inspection team was well satisfied with the level of effort, thoroughness of review, and documentation of the IDA.

At the time of the October 1988 NRC IDA team inspection, Bechtel had responded to only a few of the action items and SWEC had evaluated only a few of those responses. Another inspection addressing these aspects of the IDA was considered appropriate. This second team implementation inspection took place during the week of January 3, 1989, at SWEC's offices in Cherry Hill, New Jersey. The results of this second inspection are discussed in Inspection Report 50-353/88-203. The inspection team assessed the adequacy of SWEC's closure of design action items based on the Bechtel response and SWEC's evaluation. Also, this inspection assessed whether the sampled observation reports properly included all the unresolved issues and the associated action items. The team also reviewed the status of previous IDA inspection report findings and SWEC's resolution of two design-related issues identified by the NRC ICA inspection team members.

The IDA inspection team members were generally satisfied with the level of effort of the IDA and closure of sampled action items. Of the 86 action items reviewed by the NRC inspection team, only 6 had issues identified where additional review might be appropriate. The need for additional review reflected a difference of opinion as to where to conclude technical review of an item or whether to include a specific technical item based on the experience of the reviewer. There were seven items identified by the NRC inspection team where PECo and SWEC were requested to provide followup information. The inspection team concluded that if the IDA is continued with the same thoroughness and attention to detail that has been observed to date, the IDA could support a conclusion regarding the design adequacy of Limerick Unit 2.

The licensee's report on the IDA is expected to be submitted the second week of April 1989. A team inspection of the results in this report is scheduled for the week of April 24, 1989, at the corporate offices of Bechtel Power Corporation in San Francisco, California. The focus of this upcoming inspection will be on resolution of the design observation reports and assessment of any associated corrective actions identified in the IDA final report.

As identified above, the NRC staff has performed an extensive review of the licensee's comprehensive IDCA. The team has not to date discovered any significant design or construction deficiency that would raise a question about the readiness of Unit 2 for an operating license.

18 HUMAN FACTORS ENGINEERING

18.2 Safety Parameter Display System

The safety parameter display system (SPDS) as described in NUREG-0737, Supplement 1, was implemented at Limerick Unit 1 using the General Electric (GE) emergency response information system (ERIS). ERIS is GE's tradename and the Limerick specific system is entitled the emergency response facilities data system (ERFDS). The staff's acceptance of the Limerick Unit 1 SPDS was discussed in SSERs 3 and 5. The designations ERIS and ERFDS were used interchangeably to discuss the same system.

As described in SSER 5, ERFDS is a stand-alone, dual computer system which includes software for the SPDS and transient analysis recording functions. Color graphic display terminals are located in the control room, technical support center (TSC), and emergency operations facility (EOF) for viewing the SPDS displays.

The SPDS for Unit 2 has been implemented using the GE plant monitoring system (PMS). The PMS includes all of the ERFDS functions described above in addition to software for NSSS core monitoring, balance of plant, and point-log-alarm. The SPDS displays for Unit 2 are identical to the displays used on Unit 1. The PMS data base for Unit 2 includes all of the parameters monitored by ERFDS on Unit 1. The same number of SPDS terminals were provided for Unit 2 in the control room, TSC, and EOF.

APPENDIX A

CHRONOLOGY
LIMERICK GENERATING STATION, UNIT 2

August 1, 1985	Generic Letter 85-14 regarding commercial storage at power reactor sites of low-level radwaste not generated by utility.
August 2, 1985	Generic Letter 85-13 transmitting NUREG-1154 regarding the event at Davis-Besse of loss of main and auxiliary feedwater.
August 6, 1985	Generic Letter 85-15 regarding information concerning deadlines for compliance with 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."
August 6, 1985	Letter from licensee forwarding and discussing revised FSAR Chapter 15.
August 7, 1985	Letter to licensee forwarding Supplement 6 to the safety evaluation report for Limerick (SSER 6, NUREG-0991).
August 21, 1985	Generic Letter 85-16 regarding high boron concentrations.
August 21, 1985	Generic Letter 85-17 regarding availability of Supplements 2 and 3 to NUREG-0933, "Prioritization of Generic Safety Issues."
September 24, 1985	Generic Letter 85-18 regarding operator licensing examinations.
September 24, 1985	Generic Letter 85-19 regarding reporting requirements on primary coolant iodine spikes.
October 3, 1985	Letter from licensee forwarding Amendment 80 to applications for Operating License NPF-39 and Construction Permit CPPR-107 regarding changing Final Safety Analysis Report (FSAR) Chapter 15 and automatic depressurization system logic drawing. Rev. 44 to FSAR also enclosed.
October 17, 1985	Letter from licensee submitting proposed schedule to satisfy requirements of final risk reduction system for anticipated transients without scram (ATWS) in accordance with Generic Letter 85-06.

October 23, 1985	Letter from licensee submitting 10-year interval inservice inspection (ISI) program plan and augmented ISI program for Unit 1.
November 7, 1985	Letter to licensee informing that earlier responses to Generic Letter 83-28, Item 1.2, "Post-Trip Review: Data and Information Capabilities," were insufficient.
November 27, 1985	Generic Letter 85-22 regarding potential for loss of recirculation capability following a loss-of-coolant accident as a result of insulation debris blockage.
December 17, 1985	Letter from licensee forwarding Rev. 45 to FSAR and Amendment 81 to application for Operating License NPF-39 and Construction Permit CPPR-107.
January 3, 1986	Generic Letter 86-01 regarding safety concerns associated with pipe breaks in boiling-water reactor (BWR) scram system.
January 6, 1986	Letter to licensee forwarding safety evaluation report (SER) in response to request for utility to review existing Limerick Technical Specifications for post-maintenance testing requirements in accordance with Generic Letter 83-28, Items 3.1.3 and 3.2.3.
January 21, 1986	Generic Letter 86-02 regarding technical resolution of Generic Issue B-19, "Thermal-Hydraulic Stability."
January 23, 1986	Letter to licensee forwarding two draft reports regarding technical insights gained from probabilistic risk assessments for information and comments.
February 10, 1986	Generic Letter 86-03 regarding applications for license amendments.
February 11, 1986	Letter from licensee requesting temporary schedular exemption to requirements to delay submittal of updated FSAR for Unit 1 until 12 months after issuance of a full-power operating license for Unit 2.
February 13, 1986	Generic Letter 86-04 regarding policy statement on engineering expertise on shift.
February 25, 1986	Letter from licensee forwarding "Summary of Validation Program for SPDS (safety parameter display system)" to close license condition 2.c.(5).
March 20, 1986	Generic Letter 86-07 with regard to San Onofre Unit 1 loss of power and water hammer event.
March 23, 1986	Generic Letter 86-08 regarding availability of Supplement 4 to NUREG-0933, "Priorization of Generic Safety Issues."

March 27, 1986	Letter from licensee informing the staff of its program regarding the use of snubbers on piping systems and requesting approval for use of ASME Code Cases N-411 and N-397.
March 31. 1986	Letter from licensee forwarding Amendment 82 to operating license application and Rev. 46 to FSAR.
March 31, 1986	Generic Letter 86-09 regarding technical resolution of Generic Issue B-59-(N-1) loop operation in boiling-water reactors and pressurized-water reactors.
April 1, 1986	Letter from licensee submitting plans to update FSAR commitment for ASME Section XI Code Edition to be in effect during preservice inspection.
April 23, 1986	Letter to licensee forwarding the safety evaluation report regarding the responses to Generic Letter 83-28, Item 1.2.
April 24, 1986	Generic Letter 86-10 regarding implementation of fire protection requirements.
April 25, 1986	Letter from licensee requesting NRC approval of application for alternative approach to postulating arbitrary intermediate pipe breaks.
April 28, 1986	Letter from licensee responding to Generic Letter 85-11 regarding NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."
May 6, 1986	Letter to licensee requesting review of the enclosed list of plant-specific multiplant actions by May 31, 1986.
May 8, 1986	Summary of meeting on April 30, 1986, with licensee in Bethesda, Maryland, to discuss processes to be used to review issues raised in remainder of operating license review.
May 20, 1986	Letter to licensee advising that NRC approval is not needed to conduct preservice inspection per ASME Code, Section XI, 1980 Edition through Winter 1981 Addenda.
May 20, 1986	Letter to licensee requesting response to enclosed generic evaluation of BWR Owners Group position on NUREG-0737, Item II.E.4.2(7) concerning containment isolation dependability, per Section 6.2.4 of SSER 6.
May 27, 1986	Letter to licensee discussing request for schedular exemption from 10 CFR 50.71(e) to submit updated FSAR within 24 months of issuance of an operating license.

May 30, 1986	Letter to licensee accepting use of damping values in ASME Code Case N-411 and special shifting criteria in ASME Code Case N-397 as alternatives to FSAR methods, with listed limitations.
May 31, 1986	Letter from licensee forwarding marked-up list of multiplant items and respective status.
June 20, 1986	Letter from licensee confirming that generic safety evaluation, "Evaluation of Radiological Consequences for Accidental Releases Through BWR 2-Inch Vent and Purge Lines," and BWR Owners Group evaluation of NUREG-0737, Item II.E.4.2 are applicable to facilities.
June 25, 1986	Generic Letter 86-11 regarding distribution of products irradiated in research reactors.
June 27, 1986	Letter to licensee informing that proposal to eliminate arbitrary intermediate pipes (AIPs) from plant design is acceptable and justified. Approval is limited to elimination of AIPs for pipe rupture protective devices only.
July 3, 1986	Generic Letter 86-12 regarding criteria for unique-purpose exemption from conversion from use of highly enriched uranium fuel.
July 15, 1986	Letter to licensee advising that its submittal regarding nuclear employee data system (NEDS) commitments meets American Nuclear Standards Institute Standard 3.3 and is acceptable.
July 28, 1986	Letter from licensee supplementing its February 11, 1986, request for schedular exemption from 10 CFR 50.71(e) requirement to submit updated FSAR. Existing FSAR will be kept current by periodic amendments during exemption period.
July 29, 1986	Letter from licensee advising that licensee intends to complete valve testing and submit final report in accordance with IE Bulletin 85-03.
August 14, 1986	Letter from licensee providing additional information regarding the ISI and augmented ISI program plans.
August 20, 1986	Generic Letter 86-14 regarding operator licensing examinations.
August 22, 1986	Letter to licensee forwarding SER supporting response of June 20, 1986, to SSER 6 open issue regarding NUREG-0737, Item II.E.4.2(7), "Containment Isolation Dependability." Omission of containment radiation isolation signals from 2-inch vent and purge lines is acceptable.

September 22, 1986	Generic Letter 86-15 regarding information on compliance with 10 CFR 50.49.
October 3, 1986	Letter to licensee forwarding letter of September 2, 1986, from the Federal Emergency Management Agency (FEMA) responding to request for assistance in verifying satisfactory resolution of license condition on traffic control for emergency planning. Requests coordination of planning efforts with offsite authorities.
October 16, 1986	Summary of meeting held on October 8, 1986, with licensee, General Electric (GE), and Bechtel in Bethesda, Maryland, regarding use of ASME Code Case N-411 in ongoing facility construction.
October 17, 1986	Generic Letter 86-17 regarding availability of NUREG-1169, "Technical Findings Related to Generic Issue C-8; Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods."
October 31, 1986	Letter from licensee forwarding Amendment 83 to applications for operating licenses, Rev. 14 to emergency plan and Rev. 47 to FSAR.
November 13, 1986	Summary of meeting held on October 1, 1986, with licensee in Bethesda, Maryland, regarding schedules for licensing actions.
November 21, 1986	Summary of meeting held on November 1, 1986, with licensee, GE, and Bechtel in Bethesda, Maryland, regarding use of ASME Code Case N-411 concerning alternative damping values for seismic analysis of piping Classes 1, 2, and 3.
December 1, 1986	Letter from licensee forwarding proposed revisions to physical security plan in accordance with amendment of August 4, 1986, to 10 CFR 73.55.
December 18, 1986	Letter from licensee forwarding Amendment 84 to application for operating license and Rev. 8 to "Fire Protection Evaluation Report."
December 24, 1986	Letter from licensee forwarding documentation confirming information presented at the meeting of December 3, 1986, regarding use of ASME Code Case N-411. The information provides a basis for similarity determination of listed piping systems/lines inside containment.
December 29, 1986	Letter from licensee discussing proposal to change design and operation of reactor enclosure recirculation system. NRC review and approval of elimination of charcoal adsorber cooldown mode requested by April 1987.

January 2, 1987	Letter from licensee providing additional information to support November 17, 1986, application for amendment to Operating License NPF-39 regarding operation with partial feedwater heating and increased core flow.
January 8, 1987	Generic Letter 87-01 regarding public availability of NRC operator licensing examination question bank.
January 13, 1987	Letter from licensee forwarding information demonstrating acceptability of ASME Code Case N-411 calculations for safety/relief valve discharge lines off main steam lines and for annulus pressurization load for lower portion of head spray line.
January 13, 1987	Letter from licensee requesting changes to Technical Specifications to reflect modifications to the standby gas treatment system (SGTS) as required by License Condition 2.C.14.
January 13, 1987	Letter from licensee forwarding electrical, piping and instrumentation, and control and instrumentation drawings listed in Tables 1.7-1 through 1.7-3 of Rev. 47 to the FSAR.
January 23, 1987	Letter to licensee forwarding request for additional information regarding submittal of October 17, 1985, implementation of 10 CFR 50.62 ATWS rule.
January 26, 1987	Letter from licensee forwarding and documenting additional information regarding analytical evaluations performed to demonstrate acceptability of use of ASME Code Case ISM(ABS)/N-411, including reason for difference in response spectra grouping for snubbers.
January 30, 1987	Letter from licensee forwarding application for amendment to Operating License NPF-39, changing Technical Specifications to increase allowable control room air leakage rate.
January 30, 1987	Letter from licensee forwarding Rev. 2 to first 10-year interval inservice inspection program plan and first 10-year interval augmented inservice inspection program plan.
February 5, 1987	Letter from licensee forwarding revised emergency plan implementing procedures, including Rev. 2 to EP-102, App. 1, "Unusual Event Notification Forms," and Rev. 2 to EP-103, App. 1, "Alert Notification Forms."
February 12, 1987	Letter from licensee forwarding additional information regarding specific parameters and general methodology used in calculations to quantify potential radioactive decay heat buildup in reactor enclosure recirculation system.

February 12, 1987	Letter from licensee informing staff of progress made toward implementation of integrated scheduling, as followup to original response to Generic Letter 85-07.
February 13, 1987	Letter from licensee forwarding January 1987 revision to physical security plan, reflecting changes in corporate organization, security organization, and security hardware.
February 19, 1987	Generic Letter 87-02 regarding verification of seismic adequacy of mechanical and electrical equipment in operating reactors against seismic criteria defined in the Unresolved Safety Issue (USI) A-46 technical resolution.
February 26, 1987	Letter from licensee forwarding Amendment 85 to operating license application and Rev. 48 to FSAR.
February 27, 1987	Generic Letter 87-03 transmitting further information regarding seismic adequacy of equipment under provisions of USI A-46.
March 2, 1987	Letter from licensee forwarding errata to Rev. 48 to FSAR.
March 5, 1987	Generic Letter 87-04 regarding temporary exemption from provisions of the Federal Bureau of Investigation's (FBI's) criminal history rule for temporary workers.
March 6, 1987	Letter from licensee forwarding revisions to emergency plan implementing procedures, including Rev. 11 to EP-277, "Personnel Safety Team Phone List," and Rev. 10 to EP-278, "Security Team Phone List."
March 13, 1987	Generic Letter 87-06 regarding periodic verification of leaktight integrity of pressure isolation valves.
March 18, 1987	Letter to licensee forwarding safety evaluation supporting licensee's proposal to delete cooldown air flow path to each charcoal adsorber in reactor enclosure recirculation system filter trains.
March 19, 1987	Generic Letter 87-07 regarding final rulemaking revisions to operator licensing, 10 CFR 55, and conforming amendments.
March 23, 1987	Letter from licensee forwarding March 1987 revision to physical security plan to reflect changes to security barrier configuration and alterations to certain barriers and alarms.
March 23, 1987	Letter from licensee forwarding application for amendment to Operating License NPF-39, revising Technical Specifications surveillance requirements to include consideration of wind speed during verification of reactor enclosure secondary containment integrity.

March 25, 1987	Letter from licensee forwarding additional information regarding NPF-39 license amendment application of January 13, 1987, reflecting modifications to connect standby gas treatment system to refueling floor volume.
March 27, 1987	Letter to licensee forwarding safety evaluation of licensee's proposals to use ASME Code Case N-411 with regard to alternative damping values for seismic analysis for Classes 1, 2, and 3 piping sections.
March 27, 1987	Letter from licensee submitting additional information to license amendment application of January 30, 1987, regarding increased allowable control room leakage during operation of control room emergency fresh air supply system.
April 1, 1987	Letter to licensee advising that financial information submitted satisfies requirements of 10 CFR 140.21.
April 3, 1987	Letter from licensee forwarding application for amendment to permit operation in the second cycle.
April 6, 1987	Summary of meeting on January 15, 1987, with licensee in Bethesda, Maryland, with regard to ASME Code Case N-411.
April 23, 1987	Letter from licensee providing additional information regarding compliance with 10 CFR 50.62, "ATWS Events," per NRC request of January 23, 1987.
April 27, 1987	Letter from licensee forwarding emergency plan implementing procedures, including Rev. 10 to EP-252, "Search and Rescue/First Aid Group," and Rev. 12 to EP-277, "Personnel Safety Team Phone List."
May 11, 1987	Generic Letter 87-08 regarding implementation of 10 CFR 73.55.
May 12, 1987	Letter from licensee forwarding public version of emergency plan implementing procedures EP-204, Rev. 0, regarding mobile counting facility activation and EP-211, Rev. 3, regarding field survey group.
May 13, 1987	Letter from licensee submitting additional information regarding NPF-39 license amendment application of January 30, 1987, concerning allowable control room leakage during operation of control room emergency fresh air supply.
May 20, 1987	Letter from licensee requesting that changes discussed in license amendment application of January 30, 1987, regarding increase in allowable control room leakage during operation of control room emergency fresh air supply be added.

June 10, 1987	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 4 to App. 1 to EP-102, "Form 1: Unusual Event Notification Message," and Rev. 9 to EP-276, "Fire and Damage Team...."
June 12, 1987	Generic Letter 87-10 regarding implementation of 10 CFR 73.57 requirements for FBI criminal history checks.
June 19, 1987	Generic Letter 87-11 regarding relaxation of the requirements for arbitrary intermediate pipe rupture.
July 9, 1987	Generic Letter 87-12 regarding loss of residual heat removal while reactor coolant system is partially filled.
August 4, 1987	Generic Letter 87-14 regarding operator licensing examinations.
August 11, 1987	Letter from licensee submitting Revision 3 to the ISI and augmented ISI program plans.
August 13, 1987	Letter from licensee forwarding application for amendment to the construction permit, CPPR-107, extending earliest and latest completion dates to May 1, 1989 and January 1, 1992, respectively.
August 27, 1987	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 5 to App. 1 of EP-102, "Unusual Event Notification Message," and Rev. 8 to EP-110, "Personnel Assembly...."
September 10, 1987	Letter from licensee forwarding Rev. 1 to PECO Document 8031-P-504, "Limerick Generating Station Preservice Inspection Program," dated August 8, 1987. Preservice examination requirements of Section XI of ASME Boiler and Pressure Vessel Code and plant-specific requests for relief identified.
September 30, 1987	Letter to licensee acknowledging receipt of letter dated December 1, 1986, transmitting proposed amendment to plant physical security plan and requesting additional information and revision to plan before NRC approval.
October 29, 1987	Letter from licensee transmitting Amendment 87 to operating license application, Rev. 50 to FSAR, and Rev. 9 to "Fire Protection Evaluation Report."
November 3, 1987	Letter to licensee forwarding safety evaluation accepting licensee's proposed "Requirements for Reduction of Risk From ATWS Events for Light-Water-Cooled Nuclear Power Plants" (10 CFR 50.62).
November 4, 1987	Generic Letter 87-15 regarding policy statement on deferred plants.

November 5, 1987	Letter from licensee forwarding plant-specific response to Item 4.5.3 of Generic Letter 83-28, including proprietary GE report, MDE-93-0485-1, "Technical Specification Improvement Analysis for Reactor Protection System for Limerick Generating Station, Units 1 and 2."
November 12, 1987	Letter from licensee requesting that reference to Rev. 7 to the fire protection evaluation report be changed to Rev. 9 in license Amendment 87.
November 12, 1987	Generic Letter 87-16 regarding transmittal of NUREG-1262, "Answers to Questions at Public Meetings Regarding Implementation of 10 CFR 55 on Operators' Licenses."
November 23, 1987	Letter to licensee forwarding safety evaluation and technical evaluation report regarding offsite dose calculation manual, Rev. 4. Problems with liquid flows and gaseous effluent pathways noted.
December 7, 1987	Letter from licensee forwarding December 1987 proposed revisions to physical security plan in response to NRC comments of September 30, 1987.
December 8, 1987	Letter to licensee discussing updated FSAR and commitment to submit changes to FSAR periodically, including revisions necessary to reflect changes made per 10 CFR 50.59.
December 17, 1987	Letter to licensee issuing Special Nuclear Materials License, SNM-1977
January 13, 1988	Letter from licensee forwarding public version of revised emergency plan implementing procedures, including Rev. 14 to EP-277, "Personnel Safety Team Phone List," and Rev. 3 to EP-313, "Distribution of Thyroid...."
January 20, 1988	Generic Letter 88-02 regarding Integrated Safety Assessment Program II (ISAP II).
January 25, 1988	Generic Letter 88-01 regarding NRC position on intergranular stress corrosion cracking (IGSCC) in austenitic stainless steel piping of boiling-water reactors.
January 27, 1988	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 12 to EP-279, "Emergency Operations Facility Group Phone List," and Rev. 14 to EP-280, "Technical Support Center...."
January 29, 1988	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 13 to EP-278, "Security Team Phone List," and Rev. 12 to EP-292, "Chemistry Sampling and Analysis Team Phone List."

February 18, 1988	Letter from licensee forwarding revision to physical security plan of page changes dated February 1988 to reflect relocation of a portion of the security barrier to accommodate certain construction activity.
February 19, 1988	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 12 to EP-102, "Unusual Event Response," and Rev. 15 to EP-103, "Alert Response."
February 26, 1988	Letter from licensee submitting Amendment 88 to license application and Rev. 51 to FSAR.
March 4, 1988	Letter to licensee forwarding the staff's safety evaluation of the licensee's ISI and augmented ISI program plans for Unit 1.
March 4, 1988	Letter to licensee forwarding request for additional information regarding Items 2.1 and 2.2 of Generic Letter 83-28 concerning equipment classification and vendor interface on reactor trip system components.
March 11, 1988	Letter to licensee forwarding milestones chart established for development of Technical Specifications for Unit 2.
March 17, 1988	Generic Letter 88-05 regarding boric acid corrosion of carbon steel reactor pressure boundary components in pressurized water reactors.
March 17, 1988	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 4 to EP-254, "Vehicle and Evacuee Control Group," and Rev. 3 to EP-255, "Vehicle Decontamination."
March 22, 1988	Generic Letter 88-06 regarding removal of organization charts from administrative control requirements of the Technical Specifications.
March 29, 1988	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 4 to EP-260, "Fire and Damage Control Team," and Rev. 4 to EP-313, "Distribution of Thyroid Blocking Tablets."
March 30, 1988	Letter from licensee forwarding Rev. 5 to offsite dose calculation manual, per staff request of November 23, 1987.
April 5, 1988	Letter from licensee forwarding Amendment 89 to license application, revising fire protection evaluation report by making various changes to accurately reflect facility systems, including modifications, and providing partial description of plant fire protection features.
April 7, 1988	Generic Letter 88-07 regarding modified enforcement policy with regard to 10 CFR 50.49.

April 8, 1988	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 4 to EP-211, "Field Survey Group," and Rev. 7 to EP-230, "Chemistry Sampling and Analysis Team."
April 8, 1988	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 15 to EP-279, "Emergency Operations Facility Group Phone List," and Rev. 17 to EP-280. "Technical Support Center...."
April 11, 1988	Letter from licensee transmitting Readiness Program Assessment Report dated March 8, 1988.
April 28, 1988	Letter from licensee forwarding revised emergency plan implementing procedures, including Rev. 6. to EP-120, "Emergency Response Manager," and Rev. 7 to EP-203, "Emergency Operations Facility Activation."
May 3, 1988	Letter to licensee forwarding order extending construction completion dates for facility to May 1, 1989 and January 1, 1992, respectively, in response to August 13, 1987 application.
May 6, 1988	Letter from licensee forwarding Rev. 14 to Emergency Plan Implementing Procedure EP-292, "Chemistry Sampling and Analysis Team Phone List," per 10 CFR 50, App. E, and revised index.
May 6, 1988	Letter from licensee for Amendment of Material License SNM-1977.
May 6, 1988	Letter from licensee providing an outline of the proposed independent design and construction assessment.
May 10, 1988	Letter to licensee forwarding request for additional information regarding facility preservice inspection program.
May 17, 1988	Generic Letter 88-09 regarding operators license under 10 CFR 55, pilot testing of fundamentals examination.
May 27, 1988	Letter from licensee forwarding revisions to physical security plan.
June 1, 1988	Letter from licensee forwarding the independent design and construction assessment program of Limerick Unit 2 for review and approval.
June 2, 1988	Letter to licensee requesting updated information supporting NRC operating license antitrust review.
June 3, 1988	Letter from licensee forwarding proposed Technical Specifications draft developed for Unit 2.

June 20, 1988	Letter from licensee forwarding State of Pennsylvania approval for plant to use new cooling water additive and to discharge Unit 2 rinses and flushes with chemically treated water through holding pond.
June 21, 1988	Letter from licensee forwarding FSAR changes required to implement accelerated test program and technical justifications for changes.
July 1, 1988	Letter from licensee forwarding Amendment 90 to license application and Rev. 52 to the FSAR to reflect plant design and incorporate revised FSAR figures.
July 1, 1988	Generic Letter 88-10 regarding purchase of security containers approved by the General Services Administration.
July 7, 1988	Letter from licensee forwarding Rev. 1 to independent design and construction assessment program for Limerick Unit 2.
July 12, 1988	Generic Letter 88-11 regarding NRC position on radiation embrittlement of reactor vessel materials and impact on plant operations.
July 26, 1988	Letter from licensee responding to NRC Bulletin 87-02, Supplements 1 and 2, "Fastener Testing to Determine Conformance With Applicable Material Specifications."
July 28, 1988	Letter to licensee notifying that submittal of Rev. 1 to program for independent design and construction assessment (IDCA) for Unit 2 is acceptable.
July 29, 1988	Letter from licensee forwarding revision to physical security plan to reflect inclusion of Unit 2 in protected area boundary scheduled to become effective on March 1, 1989.
August 1, 1988	Letter from licensee forwarding Amendment 91 to license application and Rev. 53 to FSAR to accurately reflect corporate and onsite management as of June 1, 1988.
August 2, 1988	Generic Letter 88-12 regarding removal of fire protection requirements from Technical Specifications.
August 2, 1988	Letter from licensee responding to NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping."
August 3, 1988	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 13 to EP-102, "Unusual Event Response"; Rev. 16 to EP-103, "Alert Response"; and Rev. 15 to EP-104, "Site Emergency Response."

August 8, 1988	Generic Letter 88-13 regarding operator licensing examinations.
August 8, 1988	Generic Letter 88-14 regarding instrument air supply system problems affecting safety-related equipment.
August 15, 1988	Letter from licensee forwarding responses and schedules for submittals to NRC request of May 10, 1988, for additional information regarding preservice inspection examination plan. PECo Document 8031-P-505, Rev. 5 dated April 26, 1988, included.
August 15, 1988	Letter from licensee forwarding updated information for antitrust review per request of June 2, 1988.
August 16, 1988	Letter from licensee advising of licensee's intent to comply with Section 4.2.4.3 of the SER by using GE post-irradiation fuel surveillance program.
August 19, 1988	Letter from licensee forwarding turbine system maintenance program for NRC approval.
August 31, 1988	Letter from licensee forwarding proposed operator licensing examination schedule per Generic Letter 88-13.
September 12, 1988	Generic Letter 88-15 regarding electric power system - inadequate control over design processes.
September 16, 1988	Letter from licensee forwarding the readiness verification program report.
September 16, 1988	Letter from licensee forwarding response to Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems."
September 23, 1988	Letter from licensee forwarding request for exemption from 10 CFR 50, Appendix J, Paragraphs II.H.4 and III.C regarding suppression inboard containment isolation valves located on lines that terminate below minimum suppression pool water level.
October 4, 1988	Generic Letter 88-16 regarding removal of cycle-specific parameter limits from Technical Specifications.
October 4, 1988	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 11 to EP-110, "Personnel Assembly and Accountability," and Rev. 9 to EP-208, "Security Team."
October 17, 1988	Generic Letter 88-17 regarding loss of decay heat removal.
October 20, 1988	Generic Letter 88-18 regarding storing plant records on optical disks.

October 21, 1988	Letter from licensee forwarding revised emergency plan implementing procedures: Rev. 5 to EP-211, "Field Surve Group," and Rev. 4 to EP-251, "Plant Survey Group."
October 27, 1988	Letter from licensee forwarding Rev. 1 to "Limerick Generating Station Unit 2 Reactor Pressure Vessel Preservice Inspection Examination Plan."
October 27, 1988	Letter from licensee discussing detailed control room design review.
October 28, 1988	Generic Letter 88-19 regarding use of deadly force by licensee guards to prevent theft of special nuclear material.
October 28, 1988	Letter from licensee providing supplemental technical information regarding special nuclear material license amendment request for Limerick Unit 2.
November 1, 1988	Letter from licensee forwarding Amendment 92 to operating license application and Revisions 54 and 55 to FSAR.
November 9, 1988	Letter from licensee proposing to remove rod sequence control system and to lower rod worth minimizer low-power set point from 20 to 10 percent.
November 23, 1988	Generic Letter 88-20 regarding individual plant examinations for severe accident vulnerabilities.
November 23, 1988	Letter from licensee responding to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Impact on Plant Operations."
November 23, 1988	Letter from licensee forwarding Rev. 0 to "Pump and Valve Inservice Testing Program, First 10-Year Interval."
November 29, 1988	Letter from licensee forwarding Stone & Webster Engineering Corporation letter of October 28, 1988 providing summary description of independent design construction assessment.
December 2, 1988	Letter from licensee forwarding revisions to emergency plan implementing procedures: Rev. 6 to EP-106, Rev. 7 to EP-120, and Rev. 4 to EP-244.
December 5, 1988	Letter from licensee requesting permission to complete safety parameter display system and emergency response facility display system verification during power ascension test program and to declare plant data system operable within 30 days of completion of 100-hour warranty run.

December 5, 1988	Letter from licensee requesting exemption from 10 CFR 50.44(c)(3)(i) to extend time of operation with non-inerted containment to accommodate completion of power ascension test program, justifications included.
December 6, 1988	Letter to licensee forwarding request for additional information regarding the licensee's intention to combine emergency operations facilities for plants at single Coatsville service building facility.
December 7, 1988	Letter to licensee forwarding Amendment 12 to License NPF-39 and safety evaluation.
December 9, 1988	Letter to licensee advising that methodology described in Rev. 5 to offsite dose calculation manual is within guidelines of NUREG-0133 and acceptable.
December 9, 1988	Letter from licensee forwarding public version of revised emergency plan implementing procedures, including Rev. 14 to EP-102, "Unusual Event Response," Rev. 17 to EP-103, "Alert Response," and Rev. 16 to EP-104, "Site Emergency Response."
December 14, 1988	Letter from licensee transmitting application for amendment to License NPF-39, consisting of Technical Specification changes to rod sequence control system and rod worth minimizer low-power set point.
December 14, 1988	Letter from licensee forwarding public version of emergency plan implementing procedures: Rev. 7 to EP-210, Rev. 16 to EP-277, Rev. 1 to EP-320, Rev. 0 to EP-321 and EP-322, and Appendix 9 to EP-322.
December 16, 1988	Letter from licensee stating that fracture toughness analysis of pressure vessel will be submitted by April 28, 1989.
December 22, 1988	Letter from licensee forwarding marked-up FSAR pages to clarify areas of electrical power distribution systems for common systems, automatic diesel generator start signals and bypassing of diesel generator trip signals.
December 29, 1988	Letter from licensee forwarding Amendment 3 to Indemnity Agreement B-101.
December 30, 1988	Letter from licensee forwarding documentation to support NRC inspection to determine compliance with RG 1.97.
January 3, 1989	Letter from licensee advising that inservice inspection plan for Unit 2 will be submitted within 1 year of date

	of issuance of an operating license. Inservice inspection program augmentation for outboard feedwater check valves will be addressed in the inservice inspection plan.
January 3, 1989	Letter from licensee forwarding Rev. 2 to PECo Document 8031-P-504, "Preservice Inspection Program."
January 6, 1989	Letter from licensee advising that Peach Bottom and Limerick annual emergency preparedness exercises are scheduled for June 7 and November 16, 1989, respectively.
January 10, 1989	Letter to licensee forwarding Amendment 13 to License NPF-39, adding new valves and controls to existing list of containment isolation valves and deleting Note 28 from Technical Specification Table 3.6.3-1.
January 11, 1989	Letter to licensee forwarding Amendment 14 to License NPF-39 changing Technical Specifications to delete primary containment isolation valves and instrumentation associated with permanent removal of reactor vessel head spray piping.
January 12, 1989	Letter from licensee forwarding QA program change summary list to aid in review of revised program description in Section 17.2 of the FSAR.
January 18, 1989	Letter to licensee forwarding Amendment 15 to License NPF-39 changing Technical Specifications to reflect proposed modifications to certain containment penetrations.
January 23, 1989	Letter from licensee forwarding application for amendment to License NPF-39, consisting of Technical Specification Change Request 88-12, permitting increased pore size of filters to determine level of particulate contamination present in emergency diesel generator fuel oil.
January 23, 1989	Letter to licensee forwarding Independent Construction Assessment Inspection Report 50-353/88-202.
January 23, 1989	Letter from licensee submitting annual report of challenges to safety/relief valves for 1988.
January 27, 1989	Letter from licensee submitting application for amendment to License NPF-39 including Technical Specification Change Request 88-15 for effluent offsite dose limits.
January 27, 1989	Letter from licensee submitting application for amendment to License NPF-39 including Technical Specification Change Request 88-06 allowing inclusion of Unit 2 equipment that will be relied on to support operation of Unit 1 when Unit 2 is issued an operating license.

January 27, 1989	Letter from licensee submitting application for amendment to License NPF-39 including Technical Specification Change Request 88-07 to accommodate second refueling of reactor with new, previously irradiated, and reconstituted fuel bundles.
January 30, 1989	Letter from licensee forwarding NPDES (national pollution discharge elimination system) permit for plant covering waste water discharges associated with full operation of Units 1 and 2.
January 30, 1989	Letter from licensee responding to NRC letter of November 30, 1988, regarding concern about the ability of engineering to support facility (noted in Inspection Report 50-352/88-20).
January 31, 1989	Generic Letter 89-01 regarding implementation of programmatic controls for radiological effluent technical specifications in administrative controls section of Technical Specifications.
February 2, 1989	Letter from licensee providing revised dates for facility emergency preparedness exercises in 1989.
February 6, 1989	Letter from licensee submitting information regarding current position and plans for shift technical advisor (STA), per Generic Letter 86-04.
February 6, 1989	Letter from licensee discussing changes to Unit 2 FSAR since Unit 1 licensing changes reflect modifications made to Unit 1 under 10 CFR 50.59 provisions and incorporated into Unit 2 design and design changes.
February 7, 1989	Letter to licensee forwarding safety evaluation approving deletion of rod sequence control systems and lowering the rod worth minimizer low power set point to 10 percent.
February 7, 1989	Letter from licensee requesting exemption from 10 CFR 50.33(k)(1) to extend required time for submittal of report indicating how reasonable assurance will be provided that funds will be available to decommission facility.
February 7, 1989	Letter from licensee forwarding public version of revised emergency plan implementing procedure (EPIP), including Rev. 15 to EP-231, Rev. 11 to EP-237, Rev. 12 to EP-252, and Rev. 16 to EP-291.
February 10, 1989	Letter from licensee forwarding the independent construction assessment report per Inspection Report 50-353/88-202.
February 15, 1989	Letter to licensee forwarding draft Technical Specifications for review and comments.

February 16, 1989	Letter from licensee informing NRC that listed changes were made to plant emergency plan.
February 17, 1989	Letter from licensee forwarding engineering report EDE-04-0189, "Controls Systems Common Failure Analyses Report Covering Limerick Generating Station, Unit 2."
March 22, 1989	Letter from licensee regarding initial inspection of the reload-1 fuel assemblies in the pool that were the most likely candidates for reinsertion.
April 3, 1989	Letter from licensee forwarding the results of the evaluation of crud-induced localized corrosion (CILC) from reload-1 fuel assembly inspection and application of lessons learned to Units 1 and 2.
April 3, 1989	Letter from licensee forwarding evaluation of an indication of stress-corrosion cracking discovered in the recirculation riser nozzle to safe-end weld.

Appendix H

Principal Staff Contributors

Supplement 7 to the SER is a product of the NRC staff. The NRC staff members listed below were principal contributors to this report.

Name	Unit
Raj K. Anand	Plant Systems
Walter R. Butler	Project Directorate I-2
Richard J. Clark	Project Directorate I-2
Roy L. Fuhrmeister	Resident Inspector
Robert A. Gramm	Senior Resident Inspector
Robert A. Hermann	Materials Engineering
Eugene V. Imbro	Special Inspections
George Johnson	Materials Engineering
Eugene M. Kelly	Senior Resident Inspector
Thomas J. Kenny	Senior Resident Inspector
James C. Linville	Projects No. 2, RI
Robert E. Martin	Project Directorate I-2
Charles E. Morris	Electrical Engineering
Margaret B. O'Brien	Project Directorate I-2
Ronald W. Parkhill	Special Inspections
Larry L. Scholl	Resident Inspector
Steven R. Stein	Special Inspections
Frank J. Witt	Chemical Engineering

APPENDIX R

UPDATED STAFF RESPONSE TO ACRS REPORT OF NOVEMBER 6, 1984

Appendix J to SSER 4 contained the Advisory Committee for Reactor Safeguards (ACRS) Report of November 6, 1984, on the application by Philadelphia Electric Company for a full-power license for Limerick Unit 1. The Committee recommended 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.
- In view of the demography of the Limerick site, we recommend that Limerick receive special attention in the NRC staff's consideration of USI A-17, "Systems Interactions in Nuclear Power Plants."

The staff's response to the above recommendations is provided below.

The Seismic Design Margins Program was initiated by the NRC's Office of Research in 1984 to answer the need identified by the ACRS to "quantify seismic margins." Seismic margins review procedures were developed to give an effective and efficient way to assess the capability of nuclear power plants to safely withstand earthquakes larger than their design-basis level. The methodology used insights from seismic probabilistic risk assessments (PRAs) (including Limerick's PRA) and new test and earthquake experience to focus the review effort. Systems not important to mitigating core melt and components known to have high seismic strength are not analyzed in a seismic margins review. A single-level earthquake (typically 0.3 g peak ground acceleration) is used rather than PRA seismic hazard curves; this greatly simplifies the review. Rigorous plant walkdown procedures also are an important part of the review methodology.

A seismic margins review of the Maine Yankee Atomic Power Station was completed in 1987 and was the basis for a safety evaluation report resolving seismic design issues at that plant. A number of seismic vulnerabilities were found and corrected as a result of the review.

The Electric Power Research Institute (EPRI), Georgia Power, and the NRC began the cooperative seismic margins review at Hatch Unit 1 in 1988. The first major plant walkdown took place in November 1988, and the final evaluations and documentation should be completed in the summer of 1989. Georgia Power is performing its Unresolved Safety Issue (USI) A-46 review of components and tanks in conjunction with the margins review. The lessons learned from this sharing of efforts is of great interest to the NRC staff.

With the completion of the Hatch review, the Seismic Design Margins Program will have met its original objectives and thus is planned to be closed out within the next year.

However, the seismic margins approach is also currently being considered in the implementation of the Severe Accident Policy Statement. There may be further methodology developments to meet the specific needs of individual plant examination.

The resolution of USI A-17 has considered a number of actions for addressing the issue of "systems interactions." As presently proposed, the resolution relies on the continuing review of operating experience and subsuming concerns regarding flooding and water intrusion from internal plant sources into the individual plant examinations.

APPENDIX Q

SAFETY EVALUATION REPORT SUPPLEMENT
PRESERVICE INSPECTION RELIEF REQUEST EVALUATION
FOR LIMERICK UNIT 2

I. INTRODUCTION

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

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

Requests for relief from the ASME Code Section XI requirements which the licensee has determined to be impractical for systems and components at Limerick Generating Station, Unit 2, were identified in Revision 2 of the Preservice Inspection Program, dated December 27, 1988. These relief requests were supported by information pursuant to 10 CFR 50.55a(a)(3). Therefore, the staff evaluation consisted of reviewing these submittals with regard to the requirements of the applicable Code edition and addenda and determining if relief from the Code requirements was justified.

II. TECHNICAL REVIEW CONSIDERATIONS

A. The construction permit for Limerick Generating Station, Unit 2, was issued on June 19, 1974. In accordance with 10 CFR 50.55a(g)(2), components (including supports) which are classified as ASME Code Class 1 and 2 are required to be designed and provided with access to enable the performance of required preservice examinations.

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

C. The intent of a preservice examination is to establish a reference or baseline prior to the initial operation of the facility. The results of subsequent inservice examinations can then be compared with the original condition to determine whether changes have occurred. If the inservice inspection results show no change from the original condition, no action is required. In the case where baseline data are not available, all flaws must be treated as new flaws and evaluated accordingly. Section XI of the ASME Code contains acceptance standards which may be used as the basis for evaluating the acceptability of such flaws.

D. Other benefits of the preservice examination include providing redundant or alternative volumetric examination of the primary pressure boundary using a test method different from that employed during the component fabrication. Successful performance of the preservice examination also demonstrates that the welds so examined are capable of subsequent inservice examination using a similar test method.

E. In some instances where the required preservice examinations were not performed to the full extent specified by the applicable ASME Code, the staff may require that these examinations or supplemental examinations be conducted as a part of the inservice inspection program. Requiring supplemental examinations be performed during the preservice inspection would result in hardships or unusual difficulties without a compensating increase in the level of quality or safety. The performance of supplemental examinations, such as surface examinations, in areas where volumetric examination is difficult, will be more meaningful after a period of operation. Acceptable preoperational integrity has already been established by similar ASME Code Section III fabrication examinations. Failure to perform a 100% preservice examination of the welds identified below will not affect the assurance of initial structural integrity.

In cases where parts of the required examination areas cannot be effectively examined because of a combination of component design and current examination technique limitations, the development of new or improved examination techniques will continue to be evaluated. As improvements in these areas are achieved, the staff will require that these new techniques be made a part of the inservice examination requirements for the components or welds which received a limited preservice examination.

Several of the preservice inspection relief requests involve examination of less than the required volume of a specific weld. The inservice inspection program is based on the examination of a representative sample of welds to detect generic service-induced degradation. In the event that the welds identified in the preservice inspection relief requests are required to be examined again, the possibility of augmented inservice inspection will be evaluated during review of the licensee's initial 10-year inservice inspection program. An augmented program may include increasing the extent and/or frequency of examination of accessible welds.

III. EVALUATION OF RELIEF REQUESTS

The licensee requested relief from specific preservice inspection requirements in Revision 2 of the Limerick Generating Station, Unit 2, Preservice Inspection Program, dated December 27, 1988. All of these relief requests were supported by information pursuant to 10 CFR 50.55a(a)(3). Based on the information submitted by the licensee and the staff's review of the design, geometry, and materials of construction of the components, certain preservice inspections required by the ASME Boiler and Pressure Vessel Code, Section XI, have been determined to be impractical to perform. The licensee is required to demonstrate that either (i) the proposed alternative would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements of this section would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3), conclusions that these preservice requirements are impractical are justified as follows. Unless otherwise stated, references to the Code refer to the ASME Code, Section XI, 1980 Edition including Addenda through Winter 1981 (80W81).

A. Relief Requests 1 and 2, Examination Category B-A, Items B1.11 and B1.12, Reactor Pressure Vessel Shell Welds

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-A, Items B1.11 and B1.12 require a 100% volumetric examination of RPV circumferential and longitudinal shell welds as defined by Figures IWB-2500-1 and IWB-2500-2 respectively.

Code Relief Request: Relief is requested from the requirement to examine 100% of the required volume of shell circumferential weld AE and shell longitudinal weld BC. The licensee reports that 88.3% of the Code-required axial scan was completed for weld AE and 85.2% of the Code-required axial scan was completed for weld BC.

Reason for Request: Circumferential shell weld AE received a limited ultrasonic examination perpendicular to the weld axis due to opposing N3 and N12 nozzle interferences. Longitudinal shell weld BC received a limited ultrasonic examination perpendicular to the weld axis due to opposing N1 and N2 nozzle interferences.

These interferences to remote automatic ultrasonic examinations also affected manual examination techniques. The area missed is the base metal directly adjacent to the nozzles.

Staff Evaluation: The staff has reviewed the licensee's submittal and determined that the weld and required volumes immediately adjacent to the relief request areas were completely examined in accordance with ASME Section XI requirements. This, in conjunction with complete examinations of the weld metal and partial Code examinations of the base metal, provide adequate baseline examination data of the relief request areas to facilitate comparison with the results of subsequent examinations. The staff concludes that the limited Section XI examination provides an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

B. Relief Request 3, Examination Category B-D, Item B3.90, Reactor Pressure Vessel Nozzle-to-Vessel Weld

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.90 requires a 100% volumetric examination of all full penetration nozzle-to-vessel welds in the RPV as defined by Figure IWB-2500-7.

Code Relief Request: Relief is requested from the requirement to examine 100% of the Code-required volume of nozzle-to-vessel weld N4D. The licensee reports that 88.9% of the Code-required volumetric examination was completed.

Reason for Request: Nozzle-to-vessel weld N4D received a limited ultrasonic examination due to the close proximity of instrumentation nozzle NIIB. The interference to remote automatic ultrasonic examinations also affected manual examination techniques. Except for the area where the N4D and NIIB nozzles are in direct opposition, complete one-sided examinations were performed in accordance with the Code. Manual examinations were performed in the area of the interference including the relief request area between the nozzles. The required weld metal was examined with two angle beams in two directions parallel to the weld and in one direction perpendicular (transverse) to the weld.

Staff Evaluation: The staff has reviewed the licensee's submittal and determined that the weld and required volumes immediately adjacent to the relief request area were completely examined in accordance with Section XI requirements. This, in conjunction with complete one-sided examinations of the weld metal and partial Code examinations of the base metal, provides adequate baseline examination data of the relief request area to facilitate comparison with the results of subsequent examinations. The staff concludes that the limited Section XI examination provides an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

C. Relief Request 4, Examination Category B-H, Item B8.10, Reactor Pressure Vessel Integrally Welded Attachments

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-H, Item B8.10 requires 100% surface examinations of RPV integrally welded attachments as defined by Figures IWB-2500-13, -14, and -15, as applicable.

Code Relief Request: Relief is requested from the requirement to examine 100% of the required surface of the welds attaching the RPV stabilizer brackets to the RPV at 45-, 135-, 225-, and 315-degree azimuths. The licensee reports that 61.3% of the Code-required surface was examined on all four stabilizer brackets.

Relief is also requested from the requirement to examine 100% of the required surface of weld build-up integral attachment FR. The licensee reports that 75% of the Code-required surface was examined on integral attachment FR.

Reason for Request: Mirror insulation support brackets limit access for surface examination of the subject RPV stabilizer brackets.

The configuration of the RPV skirt knuckle attachment to the bottom head side plates limits access for surface examination of integral attachment FR.

Staff Evaluation: The staff has reviewed the licensee's submittal and determined that except for the surfaces covered or immediately adjacent to the mirror insulation supports, 100% of the stabilizer bracket integral attachment welds received magnetic particle surface examination from two directions. The four stabilizer brackets without mirror insulation support interferences were completely examined in accordance with Section XI requirements. VT-1 visual examinations were performed in the interference areas as an alternate examination method.

Weld build-up FR was examined completely in two directions above the RPV skirt and one direction below the skirt using magnetic particle techniques. The surface below the skirt was partially examined from a second direction in conjunction with the surface examinations of skirt knuckle integral attachment weld CG. VT-1 visual examinations were performed in the interference areas as an alternative examination method.

Based on the above, the staff concludes that the limited Section XI surface examination provides an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

D. Relief Requests 5 and 6, Examination Category B-F, Item B5.130, Class 1 Dissimilar Metal Piping Welds and Examination Category B-J, Item B9.11, Pressure Retaining Welds in Class 1 Piping

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-F, Item B5.130 (dissimilar metal piping welds), and Examination Category B-J, Item B9.11 (pressure retaining piping welds) require both 100% volumetric and surface examination of welds in piping greater than or equal to 4-inch nominal pipe size (NPS). These examinations are to be as defined by Figure IWB-2500-8.

Code Relief Request: Relief is requested from examining 100% of the required volume of the following welds due to joint configuration/component design:

Relief Request 5
Examination Category B-F, Item B5.130

System	Weld Number	Code-Required Exams Completed Circ.	Axial	Surface
RHR	DLA-212-1 FW1	100%	70%	100%
RHR	DLA-212-2 FWI	100%	70%	100%
RHR	DLA-212-3 FWI	100%	70%	100%
RHR	DLA-212-4 FWI	100%	70%	100%
RHR	DLA-210-1 FWI	100%	50%	100%
RHR	DLA-211-1 FW7	100%	60%	100%

Relief Request 6
Examination Category 8-J, Item B9.11

System	Weld Number	Code-Required Exams Completed Circ.	Axial	Surface
RHR	DCA-204-1 FWI	100%	35%	100%
RHR	DCA-204-2 FWI	100%	35%	100%
RHR	DCA-205-1 FW5	100%	75%	100%
RHR	DCA-418-1 FW5	100%	50%	100%
RHR	DCA-418-3 FW5	100%	50%	100%
FW	DLA-206-1 FW4	100%	75%	100%
RECIRC	VRR-2RD-2A WA9	100%	75%	100%
RECIRC	VRR-2RD-2B WB9	100%	75%	100%
RWCU	DCA-201-1 FW2	100%	75%	100%
RWCU	DCA-201-1 FW3	100%	75%	100%
RWCU	DCA-201-1 FW10	100%	75%	100%
RWCU	DCA-201-1 FWII	100%	75%	100%
RWCU	DCA-201-1-9 SW4	100%	75%	100%
RWCU	DCA-201-1-10 SWI	100%	75%	100%
RWCU	DCA-201-3 FW9	100%	75%	100%
RWCU	DCA-201-3 FWIO	100%	75%	100%
RWCU	DCA-201-3 FW19	100%	75%	100%
RWCU	DCA-201-3 FW20	100%	75%	100%

Reason for Request: The licensee reports that nonparallel OD and ID surfaces in certain valve and fitting designs preclude complete Code coverage of the component base material adjacent to the weld when using standard examination techniques. The licensee also reports that the required examinations are being performed to the extent practical utilizing any combination of alternative beam paths and angles or smaller transducers.

Staff Evaluation: The staff has reviewed the licensee's submittal, including the specific "Incomplete Examination Reports" for each of the welds, and concluded that, as the licensee has stated, all of the subject weld designs limit ultrasonic examination.

The licensee included a "Safety Impact Summary" for each of the welds which describes sufficient system redundancy, leak detection capability, and alternative systems which have been included in the plant design to ensure plant safety.

The integrity of the piping pressure boundary has been verified by construction code examination and testing requirements. Vendor shop welds were radiographed and liquid penetrant tested in accordance with that edition of ASME Section III in effect at the time of procurement. Field weld examinations, which also include radiography and liquid penetrant, were performed in accordance with the 1974 Edition of Section III, as modified by the Winter 1974 Addenda. The installed system was subjected to a hydrostatic pressure test of 1.25 times the system design pressure.

Based on the above, the staff concludes that the limited Section XI examination, in conjunction with the fabrication examinations, provides an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

E. Relief Request 7. Examination Category B-M-2, Item B12.50, Class 1 Valve Bodies

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-M-2, Item B12.50, requires VT-3 visual examination of the internal pressure boundary surfaces of one valve in each group of valves of the same design, manufacturing method, manufacturer, and function. The examinations are to be performed once prior to initial plant startup.

Code Relief Request: Relief is requested from performing the preservice VT-3 visual examinations for valves not requiring disassembly during plant construction.

Reason for Request: For those valves not requiring disassembly during construction, it is intended that the examinations performed by the manufacturer and the field inspection performed prior to installation serve as an acceptable alternative to the Section XI preservice examination. There are seven Class @ valve groups included in this relief request. The licensee has identified the specific groups in Table B-4 of the relief request.

The VT-3 visual examination has been performed as required for those valves requiring disassembly during construction.

Staff Evaluation: The staff has determined that disassembly of the subject valves for the sole purpose of performing preservice visual examination is not practical. The integrity of each valve's pressure boundary has been verified by construction code testing requirements. This Code Category includes both cast and forged valves that were fabricated and tested in accordance with the edition of ASME Section III in effect at the time of procurement. All cast valves and components were radiographed. Surface examinations were performed on the internal, external, and component surfaces of both cast and forged valves and were repeated for final machined surfaces. All bolting was visually examined. Hydrostatic pressure tests at 1.5 times the design pressure were performed.

The staff concludes that the construction code requirements provide an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

F. Relief Request 8, Examination Category C-A, Item CI.10, RHR Heat Exchanger Shell Circumferential Weld

Code Requirement: Section XI, Table IWC-2500-1, Examination Category C-A, Item CI.10 requires a 100% volumetric examination of shell circumferential welds at gross structural discontinuities of one heat exchanger in each group of heat exchangers of similar design, size, and service as defined by Figure IWC-2500-1.

Code Relief Request: Relief is requested from examining 100% of the Code-required volume on RHR heat exchanger weld number 2BE-205 S-G-1. The licensee reports that only 85% of the Code-required volumetric examination could be performed in the circumferential direction. The subject weld received a 100% volumetric examination in the axial direction.

Reason for Request: Complete Code coverage of the subject shell-to-flange weld is limited for parallel scans due to an interfering condition caused by the protrusion of heat exchanger bolting through the flange.

Staff Evaluation: The integrity of the heat exchanger pressure boundary has been verified by construction code examination and testing requirements. Shell plates received ultrasonic and magnetic particle examinations. The entire shell, including nozzle-to-vessel and support attachment welds, was radiographed. Magnetic particle examinations were repeated for final machined surfaces and welds. All bolting and machined surfaces were visually examined. A hydrostatic pressure test was performed at 1.5 times the design pressure.

The licensee also included a "Safety Impact Summary" which described sufficient system redundancy, leak detection capability, and alternative systems which have been included in the plant design to ensure plant safety.

Based on the above, the staff concludes that the limited Section XI examination, in conjunction with the fabrication examination, provides an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

G. Relief Request 9. Examination Category C-C, Item C3.10, RHR Heat Exchanger Integrally Welded Attachments

Code Requirement: Section XI, Table IWC-2500-1, Examination Category C-C, Item C3.10 requires a 100% surface examination of integrally welded attachments whose base material design thickness is 3/4 inch or greater for one vessel in each group of vessels of similar design, size, and service as defined by Figure IWC-2500-5.

Code Relief Request: Relief is requested from examining 100% of the Code-required surface of RHR heat exchanger tiedown brackets 2AE-205 TD-1, TD-2, TD-3, and TD-4. The licensee reports that 53% of the Code-required surface examination was performed.

Reason for Request: The licensee states that 100% of the Code-required surface examination cannot be performed due to the attachment design. The interior welds of the brackets are not accessible because of their proximity to existing building structure. Clearances between gussets are not sufficient for examination access.

Staff Evaluation: The integrity of the subject integrally welded attachments and the heat exchanger pressure boundary has been verified by construction code testing requirements.

The staff concludes that the limited Section XI examinations, in conjunction with the Section III fabrication examinations, provide an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

H. Relief Request 10, Examination Category C-C, Item C3,20. Integrally Welded Attachments to Class 2 Piping

Code Requirement: Section XI, Table IWC-2500-1, Examination Category C-C, Item C3.20 requires a 100% surface examination of Class 2 piping integrally welded attachments whose base material design thickness is 3/4 inch or greater as defined by Figure IWC-2500-5.

Code Relief Request: Relief is requested from examining 100% of the Code-required surface of lugs EBB-242-K21 FW2A through FW2H on the control rod drive scram header line. The licensee reports that the subject lugs did receive 89% of the Code-required surface examination.

Reason for Request: The licensee states that 100% of the Code-required surface examination of the subject lugs cannot be performed due to an interference with existing support steel, which is not removable.

Staff Evaluation: The integrity of the piping pressure boundary has been verified by construction code examination and testing requirements. Field weld examinations and testing, which include radiography of piping circumferential welds, magnetic particle testing of attachment welds, and hydrostatic pressure tests of the installed system, were performed in accordance with the 1974 Edition of Section III, as modified by the Winter 1974 Addenda.

The staff concludes that the Section III fabrication examinations and the limited Section XI surface examination provide an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

I. <u>Relief Request 11, Examination Category C-F, Item C5.11, Class 2 Pressure Retaining Piping Welds in Piping With 1/2-Inch and Less Nominal Wall thickness</u>

<u>Code Requirement</u>: Section XI, Table IWC-2500-1, Examination Category C-F, Item C5.11 requires a 100% surface examination of preselected circumferential welds in piping less than or equal to 1/2-inch nominal wall thickness as defined by Figure IWC-2500-7.

<u>Code Relief Request</u>: Relief is requested from examining the required surface of weld GBB-213-3-4A SW2.

<u>Reason for Request</u>: The subject weld is located within a floor penetration and is inaccessible for performing magnetic particle or liquid penetrant examinations. The licensee states that this weld is not a candidate weld for future inservice inspections and, therefore, preservice inspection would not normally be required.

<u>Staff Evaluation</u>: The integrity of the subject piping pressure boundary has been verified by construction code examination and testing requirements. Vendor shop welds were radiographed in accordance with the edition of ASME Section III in effect at the time of procurement. Field weld examinations and testing, which include radiography and hydrostatic pressure tests of the installed system, were performed in accordance with the 1974 Edition of Section III, as modified by the Winter 1974 Addenda.

It is also noted that the licensee opted to examine 100% of the total population of nonexempt Code Category C-F welds within the defined ASME Section XI boundary for preservice inspection.

As this weld is not required to be examined during inservice inspections, the staff concludes that the licensee has met or exceeded the intent of the Code for preservice. Therefore, relief is granted as requested.

J. <u>Relief Requests 12 and 13, Examination Category C-F, Items C5.21 and C5.22, Class 2 Pressure Retaining Piping Welds in Piping With Greater Than 1/2-Inch Nominal Wall Thickness</u>

<u>Code Requirement</u>: Section XI, Table IWC-2500-1, Examination Category C-F, Items C5.21 and C5.22 require both 100% surface and volumetric examinations of circumferential and longitudinal welds in Class 2 pressure retaining piping with greater than 1/2-inch nominal wall thickness as defined by Figure IWC-2500-7.

<u>Code Relief Request</u>: Relief is requested from examining 100% of the required volume of the following welds due to joint configuration/component design:

Relief Request 12
Examination Category C-F, Item C5.21

System	Weld Number	Code-Required Exams Completed Circ.	Axial	Surface
RHR	HBB-218-1 FW7	100%	87%	100%
RCIC	EBB-235-1 FWI	100%	80%	100%

Relief Request 13
Examination Category C-F, Item C5.22

System	Weld Number	Code-Required Exams Completed Circ.	Axial	Surface
MS	EBB-201-1-10 SW1 LDRI	80%	80%	100%
MS	EBB-201-1-10 SW2 LU	80%	80%	100%
MS	EBB-202-1-10 SWI LD	80%	80%	100%
MS	EBB-202-1-10 SW2 LU	80%	80%	100%
MS	EBB-203-1-10 SWI LD	80%	80%	100%
MS	EBB-203-1-10 SW2 LU	80%	80%	100%
MS	EBB-204-1-11 SWI LD	80%	80%	100%
MS	EBB-204-1-11 SW2 LU	80%	80%	100%

Reason for Request: The licensee reports that nonparallel OD and ID surfaces in certain valve and fitting designs preclude complete Code coverage of the component base material adjacent to the weld when using standard examination techniques. The licensee also reports that the required examinations are being performed to the extent practical utilizing any combination of alternate beam paths and angles or smaller transducers.

Staff Evaluation: The staff has reviewed the licensee's submittal, including the specific "Incomplete Examination Reports" for each of the welds, and concluded that, as the licensee has stated, all of the subject weld designs limit ultrasonic examination.

The licensee included a "Safety Impact Summary" for each of the welds which describes sufficient system redundancy, leak detection capability, and alternative systems that have been included in the plant design to ensure plant safety.

It is also noted that the licensee opted to examine 100% of the total population of nonexempt Examination Category C-F welds within the defined ASME Section XI boundary for preservice inspection. Therefore, the total number of Class 2 piping welds clearly exceeds the Code requirements for preservice inspection.

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

Based on the above, the staff concludes that the limited Section XI examination, in conjunction with the fabrication examinations, provide an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

K. Relief Request 14, Examination Category C-C, Item C3.30, RCIC Pump Integrally Welded Attachments

Code Requirement: Section XI, Table IWC-2500-1, Examination Category C-C, Item C3.30 requires a 100% surface examination of integrally welded attachments whose base material design thickness is 3/4-inch or greater for one pump in each group of pumps of similar design, size, and service as defined by Figure IWC-2500-5.

Code Relief Request: Relief is requested from examining 100% of the Code-required surface of the RCIC pump mounting support attachments 20P-203 PSI, PS2, PS3, and PS4. The licensee reports that 84% of the Code-required surface examination was performed.

Reason for Request: The licensee states that 100% of the Code-required surface examination cannot be performed due to the attachment design. The bottom portion of the mounting support attachment welds has been made inaccessible due to its proximity to the pedest l.

Staff Evaluation: The integrity of the subject integrally welded attachments and the RCIC pump pressure boundary has been verified by construction code testing requirements.

In the "Safety Impact Summary" the licensee described system redundancy, leak detection capability, and alternative systems which have been included in the plant design to ensure plant safety.

The staff concludes that the limited Section XI surface examination, supplemented by the Section III fabrication examinations, provides an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

L. Relief Request 15, Examination Categories F-A, F-B, and F-C, Class 1. 2. and 3 Component Supports

Code Requirement: Section XI, Table IWF-2500-1 requires VT-3 and VT-4 visual examinations of the supports of those components that are required to be examined under IWB, IWC, and IWD. The examinations will include connections to the pressure retaining component, intermediate joints, the building structure, settings, and alignments of 100% of the components and structural steel listed on the Bill-of-Materials of the support detail drawing. ASME Section XI, Paragraph IWF-2200(b) further requires these examinations be performed completely, once, following the initiation of hot functional tests.

Code Relief Request: Relief is requested from the requirement for performance of the preservice VT-3 visual examinations of all supports and the VT-4 visual examinations of spring supports after the initiation of hot functional testing. The VT-4 visual examination of snubbers will, however, be performed as required following the initiation of the hot functional tests.

Reason for Request: The licensee states that to implement a program of this magnitude following the initiation of hot functional tests would require the installation/removal of large amounts of scaffolding and insulation as well as the disassembly/assembly of surrounding components solely for the purpose of providing access to perform the required visual examinations. This would involve an unnecessarily high risk of damage to surrounding components and an impact on the Plant startup testing program.

The preservice visual examinations are being performed in conjunction with the Construction Quality Assurance Program using examination personnel who are fully qualified under ASME Section XI rules. Included are 100% of the ASME Code Class 1, 2, and 3 (Quality Group A, B, and C) component supports in systems or portions of systems not exempted by Articles IWB, IWC, or IWD of Section XI.

In addition to the visual examinations, each piping system shall be "walked down. by piping stress engineers as part of the as-built reconciliation program. These engineers will perform a visual review of the system to determine that all support clearances, physical displacements, and freedoms of motion are as designed and will permit the proper free thermal and seismic movement of the system. All deviations which would either compromise piping system performance as designed or prevent the proper free movement will be documented and corrected including the performance of new preservice baseline visual examinations as required.

Staff Evaluation: The condition of the component supports after hot functional testing is being evaluated by the implementation of a preoperational and startup testing program. During preoperational and startup testing, a dynamic testing program is being implemented by the licensee in accordance with Limerick FSAR Table 3.9-7 to verify that the piping system responds, under design conditions, in an acceptable manner as predicted by analysis. Included in this program are hot deflection testing, transient vibration testing, and steady state vibration testing. The supports on piping systems not responding within established test acceptance standards will be evaluated and reworked as required and new preservice baseline visual examinations will be performed as required. These additional testing programs provide assurance that ASME Code Class 1, 2, and 3 component supports have the adequate clearance, physical displacements, and freedom of motion, thereby maintaining plant safety and reliability.

The staff concludes that the construction/preservice visual examinations, in conjunction with the preoperational and startup testing program described above, provide an acceptable level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

M. Relief Request 16, Augmented Requirements - Circumferential and Longitudinal Butt Welds in Piping Within the "No Break" Boundaries

Augmented Requirement: Mechanical Engineering Branch Technical Position MEB 3-1 and the Limerick Generating Station FSAR Section 3.6.2.1 both require 100% volumetric examination of all of the circumferential and longitudinal pipe welds within the "no break" boundaries. The examination volume shall include the full through-wall thickness of the weld and base metal on each accessible side of the weld for a distance of 1/4 inch from the weld edge. Volumetric examinations are to be performed perpendicular and parallel to the weld axis in accordance with Appendix III, Subarticles III-4420 and III-4430, respectively.

Augmented Relief Request: Relief is requested from examining 100% of the required volume of the following welds due to joint configuration/component design:

System	Weld Number	Augmented-Required Exams Completed Circ.	Axial	Surface
FW	DLA-205-1 FWI	80%	80%	100%
FW	DLA-206-1 FW1	65%	65%	100%
HPCI	DBA-206-1-3B SW4	85%	100%	100%
RCIC	DBA-207-1 FW52	85%	100%	100%
RCIC	DBA-207-1-5A WBA	85%	85%	100%
RCIC	DBA-207-1-5A W8B	85%	85%	100%
RWCU	DCA-201-1 FW6	75%	75%	100%
RWCU	DCA-20I-1 FW7	100%	75%	100%
RWCU	DCA-201-2 FWI	50%	50%	100%
RWCU	DCB-202-1 FW1	75%	75%	100%
RWCU	DCB-202-1 FW4	100%	75%	100%
RWCU	DCC-203-1 FW25	100%	75%	N/A
FW	DBB-203-1 FW8	80%	100%	100%
FW	DBB-203-1-1B SW2	80%	100%	100%
FW	DBB-203-1 FW2	62%	62%	100%
FW	DBB-203-1 FW10R1	81%	100%	100%
FW	DBB-204-1 FW2	62%	62%	100%
FW	DBB-204-1 FW4	100%	75%	100%
FW	DBB-204-1-IA SW2	85%	100%	100%
FW	DBD-203-1 FWIRI	85%	85%	N/A
FW	DBD-203-1 FW3R1	80%	100%	N/A
FW	DBD-204-1 FWIRI	83%	83%	N/A
FW	DBD-204-1 FW3RI	80%	100%	N/A

Reason for Request: The licensee reports that nonparallel OD and ID surfaces in certain valve and fitting designs preclude complete Code coverage of the component base material adjacent to the weld when using standard examination techniques. The licensee also reports that the required examinations are being performed to the extent practical utilizing any combination of alternate beam paths and angles or smaller transducers.

Staff Evaluation: The staff has reviewed the licensee's submittal, including the specific "Incomplete Examination Reports" for each of the welds, and concluded that, as the licensee has stated, all of the subject weld designs limit ultrasonic examination.

The licensee included a "Safety Impact Summary" for each of the welds which describes sufficient system redundancy, leak detection capability, and alternative systems that have been included in the plant design to ensure plant safety.

The integrity of the piping pressure boundary has been verified by construction code examination and testing requirements. Vendor shop welds were radiographed and liquid penetrant tested in accordance with that edition of ASME Section III in effect at the time of procurement. Field weld examinations, which include radiography and liquid penetrant examinations, were performed in accordance with the 1974 Edition of Section III, as modified by the Winter 1974 Addenda. The installed system was subjected to a hydrostatic pressure test of 1.25 times the system design pressure.

Based on the above, the staff concludes that the limited augmented preservice examination, in conjunction with the fabrication examinations, provides an acceptable level of preservice structural integrity and that compliance with the specific preservice augmented requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, relief is granted as requested.

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 to perform. 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.

This staff technical evaluation has not identified any practical method by which the licensee can meet all the specific preservice inspection requirements of Section XI of the ASME Code for the existing Limerick Generating Station, Unit 2, facility. Requiring compliance with all the exact Section XI required inspections would delay the startup of the plant in order to redesign a significant number of plant systems, obtain sufficient replacement components, install the new components, and repeat the preservice examination of these components. Even after the redesign efforts, 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 staff review and evaluation, it is concluded 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

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

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Supplement No. 7

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Safety Evaluation Report related to the operation of Limerick Generating Station, Units 1 and 2.

3. LEAVE BLANK

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11a. TYPE OF REPORT
Supplement No. 7 to the Safety Evaluation Report

b. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES
Pertains to Docket Nos. 50-352 and 353.

13. ABSTRACT (200 words or less)

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 licensee) 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. Supplement 2 was issued in October 1984. Supplement 3 was issued in October 1984. Supplement 4 was issued in May 1985. Supplement 5 was issued in July 1985. Supplement 6 was issued in August 1985 and Supplement 7 issued in March 1989. This Supplement 7 addresses the major design differences between Units 1 and 2, the resolution of all issues that remained open when the Unit 1 full power license was issued, the staff's assessment regarding the application by the licensee to operate Unit 2 and the issues that require resolution before issuance of an operating license for Unit 2.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS
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