U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 84-36
84-10

Docket No. 50-352
50-353

License No. CPPR-106
CPPR-107

Priority -

Category B
A

Licensee: Philadelphia Electric Company

2301 Market Street

Philadelphia, Pennsylvania 19101

Facility Name: Limerick Generating Station

Inspection At: Limerick, Pa.

Inspection Conducted: July 1 - 31, 1984

Inspectors: S. K. Chaudhary, Senior Resident Inspector — 8/10/84 date

J. T. Wiggins, Senior Resident Inspector — 8/10/84 date

R. W. Borchardt, Reactor Engineer — 8/10/84 date

Approved by: R. M. Gallo, Chief, Reactor Projects Section 2A — 8/14/84 date

Inspection Summary: Combined Inspection Report for Inspection Conducted July 1 - 31, 1984 (Report Nos. 50-352/84-36; 50-353/84-10

Areas Inspected: Routine inspections by resident inspectors and region based reactor engineer of: followup on outstanding inspection items; followup on IE bulletins and circulars; followup on construction deficiency and 10 CFR 21 reports; TMI action plan followup; witnessing of new fuel transfer from temporary storage to refueling floor; general walkthrough inspections; witnessing of portion of work under startup work orders; and review of diesel generator preoperational test results. This inspection involved 121 hours for Unit 1, 7 hours for Unit 2 by resident inspectors, and 84 hours for Unit 1 by region-based reactor engineer.

Results: No violations were identified. However, one significant unresolved item was identified concerning the conformance of the diesel generator preoperational test program with the Regulatory Guide 1.108 Position C.2.a(9)(Paragraph 6).

Region I Form 12
(Rev. February 1982)

DETAILS

1. Persons Contacted

Philadelphia Electric Company

J. M. Corcoran, Field QA Branch Head
R. Scott, Construction Engineer
G. Leitch, Station Superintendent
J. Spencer, Director, Start-up
J. Molito, Field Engineer

Bechtel Power Corporation

W. McCullough, Project Start-up Engineer
R. Bulchis, Resident Project Engineer

General Electric Company

R. Ballou, Start-up Operations

2. Followup on Outstanding Inspection Items

1) Bulletins

a. (Closed) IEB 77-05 and 77-05A: Electrical Connector Assemblies

These IEB's described failures identified by Sandia Laboratories of certain electrical connector/cable assemblies. The licensee was requested to review all connectors in safety systems which were required to function to mitigate an accident where the accident itself could adversely affect the ability of the system to perform its safety function. General Electric reviewed this matter and determined the connectors in question were used in the Main Steam Line radiation monitors, in the Standby Liquid Control System and in the Power Generation Control Complex. The design conditions in which these systems must operate were found to not create the harsh environment needed to fail the connectors.

Further, the inspector observed that controls to assure the qualification of connector assemblies in NSSS and BOP systems have been implemented through the Environmental Qualification Program.

b. (Closed) IEB 78-14: "ASCO Pilot Solenoid Valves"

All ASCO Pilot Solenoid Valves using "Buna-N" elastomers have been replaced with ASCO valves incorporating "ethylene propylene" elastomers. This bulletin is closed.

c. (Closed) IEB 79-03 and 79-03A: Longitudinal Weld Defects in ASME SA-312 Type 304 Stainless Steel Pipe

IEB 79-03 reported a problem with longitudinally welded piping materials provided to a construction facility by Youngstown Welding and Engineering Company. The licensee was requested to report all safety-related uses for this piping at Limerick and to develop a volumetric examination program to be applied to this piping. In a May 7, 1979 letter, the licensee reported that holdings onsite included 480 feet of 12 inch pipe for use as main steam safety-relief valve downcomer piping and 60 feet of 10 inch and 60 feet of 12 inch pipe not yet assigned to a system. Further, the licensee committed to a 100% ultrasonic inspection program for this pipe.

IEB 79-03A superseded IEB 79-03 and provided new action criteria. The scope of the IEB was expanded to include SA 312 Type 304 piping material from any vendor, but limited concern to only those applications where the pipe design stresses exceed 85% of that allowable by the ASME Code. In a July 16, 1980 letter, the licensee reported that, based on the minimum allowable wall thickness for SA-312 seamwelded pipe at Limerick, all safety-related system applications would be subject to design stresses which are less than 85%. Therefore, no further licensee actions were required.

d. (Closed) IEB 79-08: Events Relevant to Boiling Water Power Reactors Identified During Three Mile Island Incident

This bulletin was provided to the licensee for informational purposes and did not require a written response. The inspector reviewed the licensee's bulletin package to ensure that the bulletin was received, evaluated and acted upon as appropriate. The licensee has adequately addressed each issue as documented in PECo memoranda dated April 18, 1984 and May 2, 1984; a Bechtel letter dated June 19, 1979; and a G.E. letter dated April 30, 1979. The licensee's training records on mitigation of core damage, water level control and instrumentation, feed water system, ECCS systems, and licensee reporting requirements were reviewed and found to be in compliance with this bulletin. This item is closed.

e. (Closed) IEB 79-10: Requalification Training Program Statistics

This bulletin required facilities with an operating license to provide statistics about the failure rate on annual requalification exams and was not applicable to this licensee.

f. (Closed) IEB 79-18: Audibility Problems Encountered on Evacuation of Personnel From High-Noise Areas

This bulletin described a situation where personnel working in a high noise area of an auxiliary building could not hear an evacuation announcement. The auxiliary building evacuation was initiated by the Shift Supervisor because of an unplanned release of radioactive noble gases. This bulletin was provided to the licensee for information only. The licensee documented their review, and actions relating to this bulletin in 1) PECo memorandum dated April 21, 1980, 2) Bechtel letter dated February 29, 1980, 3) PECo memorandum dated November 27, 1979, 4) PECo letter dated November 26, 1979 and 5) Bechtel letter dated September 4, 1979. A review was conducted of speaker locations, db outputs specific to this plant, and sound level measurements taken at Peach Bottom Atomic Power Station in high noise areas for data analysis. It was determined that the speaker output is sufficient in all areas and under all operating conditions. Since actual operating measurements cannot be made, the licensee has documented their intention to verify the effectiveness of the audible alarm system in high noise areas after the plant is in operation. This item will be followed as part of outstanding item 83-23-03.

g. (Closed) IEB 79-27: Loss of Non-Class 1E Instrumentation and Control Power Supply Bus During Operation

This IEB dealt with an event at Oconee Unit 3 which resulted in a significant loss of instrument indications and control system functions. Although the licensee was not required to respond in writing to the IEB, it performed a detailed design review of the power supplies to those safety and non-safety-related controls and instrumentation needed to achieve cold shutdown. This design review and its results were reviewed by NRR as documented in Section 7.4.2.1 of the SER. NRR opened Confirmatory Item 33 to close out their review.

The inspector determined that the station Alarm Response Cards had been written to reflect the results of the licensee's design review in this area by identifying loss of power to specific buses as a possible condition for a received alarm. Consequently, the inspector considers this IEB as closed from the standpoint of planned inspection followup. NRR will close Confirmatory Item 33 in a supplemental SER.

h. (Closed) IEB 80-06: Engineered Safety Feature (ESF) Reset Controls

This bulletin described a problem in which, upon reset of an ESF Actuation signal, certain valves and equipment failed to remain in their emergency mode. The licensee, during the license application review process, responded to NRR regarding the results of its review of system and valve logics. As documented in the FSAR in response to Question 421.7 and in Section 7.3.2.3 of the SER, the licensee identified a number of valves which would reopen following reset of an automatic containment isolation. For each of these valves, the licensee either modified the valve logic designs or justified the as-built condition. NRR accepted the licensee's evaluation. The inspector verified, by a sample inspection, that the logic modifications were being tested during preoperational test 1P59.1, Containment Isolation and Nuclear Steam Supply Shutoff System.

i. (Closed) IEB 80-08: Examination of Containment Liner Penetration Welds

At Limerick backing bars are not used for flued head to containment liner penetration welds. All joints are open butt with TIG root pass, and radiographic examination is required per Bechtel Specification P-305. Therefore, this bulletin is closed.

j. (Closed) IEB 80-11: Masonry Wall Design

The re-evaluation of concrete masonry walls at the Limerick Generating Station was done to meet the requirements of the bulletin. All walls, except one, evaluated under this program met the requirements of re-evaluation criteria. For the remaining wall, an acceptable fix was developed and carried out. This bulletin is closed.

k. (Closed) IEB 80-24: Prevention of Damage Due to Water Leakage Inside Containment

The licensee's review indicated that two closed systems and one open system supply cooling water inside containment. The Reactor Enclosure Cooling Water (RECW) System and the Drywell Chilled Water system are closed systems with head tanks. Makeup to the head tanks is a manual operation and the tanks are monitored by low level alarms. Frequent alarms/makeup operations would be indicative of a system leak inside containment. Emergency Service Water (ESW) is an open system which can serve as a backup to RECW as a cooling water supply to the recirculation pumps. This backup feature is expected to be used rarely. Additionally, the flows out of the drywell equipment and floor drain sumps are monitored by flow integrators and excessive flows are alarmed in the control room. As a result, no administrative procedure changes or design changes were identified as a result of the licensee's review of the IEB.

However, during the FSAR review process, NRC:NRR determined the need for extra isolation provisions for the RECW and DWCW systems. These provisions are being incorporated into the applicable Transient Response Implementing Procedures.

The inspector used FSAR Table 6.2-17 to verify all cooling water systems that penetrate containment were identified. No systems other than RECW, DWCW and ESW were identified.

1. (Closed) IEB 81-01: Surveillance of Mechanical Snubbers

This bulletin described problems with INC mechanical snubbers and prescribed tests and inspections of INC and other mechanical snubbers for those operating plants which used them. Selected construction facilities were also required to respond; Limerick was not identified as one. However, this IEB and a 10/17/80 letter from NRC (Tedesco) to PECo (Bauer) were reviewed by the licensee's Engineering and Startup organizations. As a result, a two phase test program was established.

During preoperational testing, the Pacific Scientific mechanical snubbers at Limerick would be inspected for proper installation and orientation and stroked to show freedom of movement. This inspection and test requirement was included in preoperational test procedures 1P100.3 A-E. During startup testing of systems whose design temperatures exceed 300°F, the snubbers would again be inspected for their adequacy to accommodate system thermal movement. This startup inspection would occur during performance of STP-17.

Subsequent periodic snubber testing would be performed as required by Technical Specification (TS) 4.7.4. The inspector reviewed the draft TS 4.7.4 and identified one item of concern. The functional testing acceptance criteria shown in draft TS 4.7.4 are those applicable to hydraulic snubbers, not mechanical snubbers. The criteria for mechanical snubbers are shown in section 4.7.5 of Standard Technical Specifications. The inspector informed the licensee of this concern on 7/4/84. The resolution of this apparent discrepancy will be documented in a subsequent inspection report (50-352/84-36-01).

m. (Closed) IEB 81-03: Flow Blockage of Cooling Water to Safety System Components by Corbicula sp. (Asiatic Clam) and Mytilus sp. (Mussel)

This bulletin was issued as a result of a flow degradation through the Arkansas Nuclear One, Unit 2 Containment Cooling Units which resulted from a buildup of clams in the piping and water coolers. This item was previously reviewed in inspection report 50-352/83-20 and left open pending the results of a 1983 licensee study. The licensee's responses dated June 4, 1981 and March 18, 1983 were reviewed in inspection report 83-20 and indicated that neither Corbicula nor Mytilus were present in the source water for LGS, although some Corbicula was present downstream. In the March 18, 1983 response, the licensee stated thata preoperational program would be started in the summer of 1983 on the Schuylkill and Delaware Rivers. The inspector reviewed a draft copy of a report that covered the period of 1979 to 1983. The report indicated that while the area where Corbicula can be found has expanded in both the Delaware and Schuylkill Rivers, there are still none present in the source water for LGS. The licensee's study will continue through 1984 and is expected to provide information on clam population density and size changes in relation to varying seasonal water conditions.

n. (Closed) IEB 82-01: Alteration of Radiographs of Welds in Piping Subassemblies

This bulletin supplied information concerning altered radiographs of welds in piping subassemblies delivered to Washington Public Power Supply System by Associated Piping and Engineering Corporation (AP&E). The alteration consisted of artificial enhancement of the ASME Code specified penetrameter 4T-Hole image accomplished by 1) touch up with a soft lead pencil, 2) scribing or scratching with a sharp object, or 3) indentation with a sharp object. The inspector reviewed the licensee's response to this bulletin dated July 23, 1982 which reported that a 100% audit of AP&E radiographic films was conducted with no abnormalities found. A region based inspector also conducted a spot check of AP&E film for artificial enhancement and did not identify any abnormalities.

o. (Closed) IEB 82-03: Stress Corrosion Cracking in Thick-Wall, Large Diameter, Stainless Steel, Recirculation System Piping at BWR Plants

The stainless steel, type 304 recirculation piping at Limerick has been replaced with type 316, low carbon stainless steel which is highly resistant to IGSCC. This bulletin, therefore, is not applicable to this site.

p. (Closed) IEB 83-02: Stress Corrosion Cracking in Large Diameter, Stainless Steel, Recirculation System Piping at BWR Plants

The stainless steel, type 304 recirculation piping at Limerick has been replaced with type 316, low carbon stainless steel which is highly resistant to IGSCC. This bulletin, therefore, is not applicable to this site.

q. (Closed) IEB 84-01: Crack in Boiling Water Reactor Mark I Containment Vent Headers

Limerick has Mark II (Over-under) containment. This bulletin, therefore, did not apply to this site, and was for information only.

2) Circulars

a. (Closed) Circular 79-11: Design/Construction Interface Problem

This circular described interface problems which had been experienced in the industry. The licensee was requested to review the adequacy of the interfaces of the designers (Bechtel and General Electric) with the constructor (Bechtel). The licensee reviewed these interfaces and found them to be adequately controlled. NRC Systematic Assessment of Licensee Performance (SALP) results have supported the licensee's conclusion.

b. (Closed) Circular 80-01: Service Advice For General Electric Induction Disc Relays

This circular identified certain G.E. relays which experienced higher than normal pick-up values due to a problem with the relay's lubricant. The licensee was advised to apply G.E. Service Advice 721-162.2 to all applicable relays. The licensee identified 446 relays covered by this circular and applied Service Advice 721-162.2 to each relay. In addition, the procedure for acceptance testing time over/under voltage relays (RT-11-04012) was verified to have a step ensuring that all relays covered by this circular are properly checked and cleaned. The inspector determined that the licensee's actions have been adequate and prompt. The licensee's actions were documented in Bechtel memoranda dated May 15, 1979, August 17, 1979 and PECo memoranda dated September 10, 1980, September 22, 1980 and October 7, 1980.

c. (Closed) Circular 80-02: Nuclear Power Plant Staff Work Hours

This item proposed limitations on maximum work hours for plant staff who perform safety-related functions. Since the issuance of this circular, the NRC has published further guidance in this area in NUREG 0737 item I.A.1.3, and NRC Generic Letter 82-12. The inspector reviewed LGS Administrative Procedure A-40 Rev. 0 titled "Procedure for Working Hour Restrictions" and found it to be in compliance with the current NRC guidance. Working hour restriction procedures were found to be acceptable in Supplement 1 to the SER section 13.1.2 with the exception that A-40 did not apply to maintenance personnel. In a September 27, 1983 letter, the licensee proposed a modification to Technical Specifications that would ensure that maintenance personnel are included in overtime restrictions. The inspector verified that this modification as approved by NRR had been incorporated into the draft technical specifications.

d. (Closed) Circular 80-21: Regulation of Refueling Crews

This circular discussed the regulatory requirements applicable to fuel handling activities. The inspector reviewed th licensee's fuel handling procedures including FH 602 "Qualification of Refueling Platform Operators", and the proposed technical specifications and determined that the licensee is in compliance with this circular. The qualification and training program for refueling crews was reviewed during inspection report 50-352/84-30 and found to be acceptable.

e. (Closed) Circular 81-08: Foundation Materials

All seismic Category I safety-related structures are founded on competent rock stratum and/or concrete backfill. Therefore this circular is not applicable to this site.

f. (Closed) Circular 80-18: 10 CFR 50.59 Safety Evaluation For Changes to Radioactive Waste Systems

The inspector reviewed procedure A-5, Safety Evaluations, and verified that it had been scoped to equally apply to <u>all</u> systems described in the FSAR by either test or drawings, that it incorporate the criteria in the IEC regarding when safety evaluations are required for maintenance activities and that it required the results of the evaluation to be documented such that both the conclusions and their supporting bases are shown. Because radioactive waste systems are described in the FSAR (ref. Chapter 11), the requirements of A-5 are applicable at Limerick.

3) Violations

a. (Closed) Violation 50-352/84-24-03:

The design bases for the main steam isolation valve leakage control system (MSIV-LCS) were not translated into design drawings. The inspector verified that isometric drawings HBB-157-1 and HBB-158-1 were revised to show the air dilution inlet pipe screens. The inspector examined design change package (DCP) 0445 and verified it provided for installation of the screens prior to fuel loading. Also, the inspector noted that the licensee reviewed a sample 5 P & ID's for safety-related systems and found no similar discrepancies.

Additionally, the inspector noted that the seismic classifications for isometrics HBB-154-1, HBB-167-1, HBB-168-1 were corrected to show them to be Seismic Class 1. The inspector also reviewed the memorandum to Bechtel Engineering's Layout and Stress Engineers to direct them to assure that the seismic category designation of isometric drawings agree with specification 8031-P-300 and with the stress calculation.

4) Unresolved Items

a. (Closed) Unresolved Item 50-352/82-13-02:

No guidance or requirements were found for adding or deleting new information on hanger drawings. A 100% review of hanger drawings in Quality Control-Welding area was initiated by the licensee, and other sections of QC were also audited for similar conditions. Any discrepancy found was corrected. The inspector has observed the working of this system over a period of time and found it acceptable. This item is resolved.

b. (Closed) Unresolved Item 50-352/84-19-03:

Startup Administrative Manual does not adequately address preoperational test results evaluation by General Electric and Bechtel. The inspector reviewed AD8.3P-1, Preoperational Test Implementation, dated 7/6/84, which incorporated in paragraph 5.5.1 the scope and extent of results review by the NSSS vendor and the A-E. This revision of AD8.3P-1 served to formally identify the results review program that had been actually implemented for the completed reviews.

c. (Closed) Unresolved Item 50-352/84-24-05: Main Steam Isolation Valve Leakage Control System (MSIV-LCS) design concerns

NRR reviewed the inspector's concern regarding reactor enclosure accessibility follow use of the MSIV-LCS. NRR will generically review this concern in connection with Task Action Plan C-8, MSIV Leakage and LCS Failures, scheduled to be completed by December 1986. The inspector had no further questions on this matter.

5) Construction Deficiency Reports

a. (Closed) CDR 83-00-13: Comsip, Inc. Containment Gas Analyzers

The licensee reported receipt of a 10 CFR 21 report from Comsip, Inc. which described a defect in the catalyst used in the Comsip models K-III and K-IV containment gas monitoring systems. Thermodynamic analyses on the catalyst showed that their useful life would decrease to 10 days following an iodine release from a design basis accident. Comsip then provided new catalyst beds with useful lives of 5 months. The inspector reviewed revision 9 to material requisition request 8031-M-235 used to obtain the new catalyst bed kits and Quality Control Inspection Report M235-QCG1-2 used to document inspection of the installation of new kits in Unit 1. The inspector had no further questions.

b. (Closed) CDR 83-00-04: Limitorque Valve Motor Operator Tandem Torque Switch Deficiency

On 7/8/83, the licensee reported a defective condition applicable to Limitorque valve motor operators on 125 valves in Unit 1. As a special design consideration, the licensee had directed Limitorque to supply motor operators with tandem torque switches instead of a single switch. Electrically, the switches were in parallel. However, the results of pre-operational testing indicated that, because of excessive mechanical play in the tandem switch assembly, one of the electrically parallel torque switch contacts would remain closed after the other has opened, allowing the motor to continue to operate past the torque limit set in the operator. Based on Quality Control Inspection Records provided for review, each Unit 1 valve has been modified by removing one of the torque switches in each assembly.

c. (Closed) CDR 84-00-04: Attachment of Q-listed Commodities to Non-Q Seismic I and IIA Installations

On 4/10/84, the licensee reported a condition which involved the attachment of Q commodities such as supports for HVAC, cable trays, conduits and small pipe to non-Q concrete forms, to non-Q monorails or to non-Q monorail support steel. On 5/10/84, the licensee notified NRC Region I that this condition was evaluated and determined to be not reportable. The inspector reviewed Bechtel Quality Assurance Management Corrective Action Report 1-35, dated 2/27/84, which described the condition, determined the organizations in Bechtel responsible for investigation and corrective action and tracked the corrective actions taken. Based on this review, the inspector noted that 70 Q-commodities were identified as being attached to non-Q installations. Further, Bechtel Project Engineering determined the causes of the condition to be confusing information contained on the civil drawings regarding the Q status of top connection materials for Q-walls and design/installation errors.

Corrective actions included documentation and disposition of each instance of Q attachments to non-Q installations, revision of the applicable civil drawings to show the as-built condition for those Unit 1 walls which had been completed and to clarify the Q-status of top connections of walls yet to be completed for Units 1 and 2, imposition of a requirement to stencil non-Q structural steel in Q facilities and retraining of the applicable Project and Field engineers.

6) Part 21 Reports

a. (Closed) Part 21 Report 80-04: William Perc Actuators

The licensee determined that William Perc HVAC actuators are not used onsite.

b. (Closed) Part 21 Report 82-01: General Electric Type HFA relay contact gap and wipe adjustments

This Part 21 report discussed potentially defective, normally closed, HFA relays supplied by G.E. The problems developed from a failure to incorporate the correct gap and wipe settings for these relays. Since HFA relays were initially manufactured with normally open contacts, the relays of concern were those which had been changed to normally closed. As a result of their reviews, neither Bechtel nor GE had used normally-closed HFA relays in safety-related systems. Additionally, in response to IEB 84-02, all GE HFA relays in Class 1E systems have been or will be replaced. Further the coil assemblies in the GE HFA's in non-Class 1E will be replaced with Century HFA coil assemblies.

3. Plant Tour

Periodically during this inspection period, the inspectors toured the Unit 1 containment, reactor enclosure, control room, diesel generator enclosures; and the Unit 2 reactor enclosure. The inspectors examined completed work and work in progress for indications of defective workmanship, nonconformance to technical requirements, and general adherence to project procedures. The inspectors reviewed drawings, specifications, procedures, and reports to assess the state of completion of the facility. Special emphasis was placed on visual examination of turned-over systems for as-installed conditions. The inspectors also witnessed portions of work in progress on the following items:

- Installation of Personnel Air Lock (SWO 59F-29)

- Hydrostatic Test of Reactor Enclosure Cooling Water supply line to the recirculation pump (SWO 14A-52)

- Movement of fuel from the storage facility to the refueling floor

- Fuel bundle receipt inspection, channeling, and placement into the spent fuel pool (see paragraph 7 of this report)

The inspectors examined the above work to verify the adequacy of quality control and health physics practices, conformance to project requirements, requisite cleanliness, and to assure the use of proper measuring and test equipment.

No violations were identified.

4. TMI Action Plan Followup

a. (Closed) Item I.A.1.1 Item 1 and 2 Shift Technical Advisor

This item required the licensee to provide training that met the lessons-learned requirements by the issuance of the fuel load license and to provide a description of the current training program. In section 13.2.2 of the SER, NRR evaluated the STA training program as acceptable. The inspector reviewed the training records of the eight STA's which included lesson plans, quizes, tests and final exams. The training records are broken down into sections covering accident analysis, chemistry, radiation protection, reactor theory, Rensselaer Polytechnic Institute courses, Limerick Systems, technical specifications, thermodynamics/fluid flow and STA training programs. The inspector verified that the licensee has implemented the STA training program described in the FSAR and the SER, and is in compliance with this item.

b. (Closed) Item I.A.2.1 Immediate Upgrading of RO and SRO Training and Qualification

This item describes the experience and educational requirements for SRO license applicants. In section I.A.2.1 of the SER, NRR reviewed the qualification program and has determined that it meets the requirements of this item. Although the licensee is not currently subject to some of the requirements of this item, their compliance will be evaluated as future SRO applications are processed. The inspector reviewed the license applications for the first two groups of SRO candidates and determined that the applicable requirements of this item were met.

c. (Closed) Item I.A.2.3 Administration of Training Programs for Licensed Operators

This item requires the operating licensee to assure that instructors who teach systems, integrated responses, transient and simulator courses demonstrate SRO qualifications and be enrolled in appropriate requalification programs. In SER section I.A.2.3 NRR concluded that the applicant's program complies with the requirements of this item. The inspector reviewed the training records of two instructors and determined that the instructor certification requirements have been satisfied. One instructor has been recently certified and one other is awaiting results of his NRC exam.

d. (Closed) Item II.B.4, Item 1 Training for Mitigating Core Damage

This item required the licensee to develop a training program prior to fuel load which will teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. In section 13.2.1 of the SER, NRR concluded that the licensee has complied with the requirements of this item. The inspector reviewed the training records of shift technical advisors, and operating personnel to verify that the required training was given. The training text is complete and well organized. A condensed version of this training is also given to managers and technicians in the Instrumentation and Control, Health Physics, and Chemistry departments.

e. (Closed) II.D.1 Items 2 and 3, Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves

This item required the licensee to conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents. In section 5.2.2 of the FSAR, NRR concluded that the pressure relief system, in conjunction with the reactor protection system, will provide adequate protection against overpressurization of the reactor coolant pressure boundary. To satisfy the testing requirements of this item, the licensee participated in the BWR Owners Group program to test safety relief valves. The test program is documented in topical report NEDE-24988 P/NEDO-24988 and its applicability to the licensee is discussed in Appendix A to that report. The test results demonstrate that the licensee's relief and safety valves will adequately perform their intended function under the expected operating conditions of design-basis transients and accidents. The licensee sent its Steam Relief Valves to Target Rock for implementation of GE SIL 196 through supplement 14 and the valves have subsequently been returned. Item 3 discusses qualification of PWR block valves and is not applicable to the licensee.

f. (Closed) Item II.D.3 Direct Indication of Relief and Safety-Valve Position

This item required the licensee to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken. It also discussed valve position indication location, alarm functions, power supplies, seismic qualification, environmental qualification, and human factors analysis. NRR found that the design of the safety and relief valve position indication system meets the requirements of this item as documented in section 7.5.2.2 of the SER. The inspector verified that the acoustical monitoring system is addressed in section 7.5 of the FSAR and in the draft of technical specifications Required equipment installation was verified through direct inspection.

g. (Closed) Item II.E.4.1 Items 2 and 3 Dedicated Hydrogen Penetrations

This item required plants using external recombiners or purge systems for post accident combustible gas control to provide containment penetration systems that are dedicated to that service only, meet redundancy and single failure requirements, and are sized to satisfy flow requirements. In section 6.2.5 of the SER, NRR stated that the combustible gas control system satisfies the design and performance requirements of 10 CFR 50.44; the provisions of RG 1.7; the requirements of GDC 41,42 and 43; and the requirements of this item, and is therefore acceptable. In a letter to NRR dated June 27, 1984, the licensee provided NRR with their proposed operational controls designed to ensure that initiation of the hydrogen ecombiner system will not create a steam bypass path. The control room copy of procedure No. S58.1.B, Revision 1 was reviewed to ensure that the operational controls were incorporated into the licensee's procedures.

h. (Closed) II.K.1 IE Bulletins and Measures to Mitigate SBLOCA's and loss of FW Accident (Items 5, 17, 20, 21, 22 and 23)

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

i. (Closed) II.K.3 Final Recommendations of B&O Task Force (Items 9, 10, 12b, 13b, 15, 22a, 22b, 24 and 27)

The following items are only applicable to Westinghouse plants and therefore are not applicable to the licensee.

II.K.3.9	PID Controller
II.K.3.10	Anticipatory Trip at High Power
II.K.3.12b	Modify Anticipatory Trip

II.K.3.13b Separation of High Pressure Coolant Injection and Reactor Core Isolation Cooling System Initiation Levels - implementation

This item required the licensee to perform analysis and make modifications dealing with initiation levels of HPCI and RCIC, and the automatic restart on low water level logic of the RCIC system. In SER section 15.9.4, NRR concluded that for Limerick, the separation of HPCI and RCIC initiation levels is unnecessary. Installation of equipment for the automatic restart of RCIC on low water level is required before NRR will issue an operating license. This modification has been made to the RCIC system. The licensee has reported completion of this item in a letter to NRR dated May 22, 1984. All actions required by this item have been satisfied.

II.K.3.15 Modify Break-Detection Logic to Prevent Spurious Isolation of High Pressure Coolant Injection and Reactor Core Isolation Cooling

This item required the licensee to modify the pipe-break-detection circuitry so that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation. The modification consists of adding a time delay to the high steam flow trip logic in the HPCI and RCIC systems. This modification has been completed on both the HPCI and RCIC systems and is documented in a licensee's letter to NRR dated May 22, 1984.

II.K.3.22(a and b) Automatic Switchover of Reactor Core Isolation Cooling System Suction - verify procedures

This item required the licensee to implement an automatic RCIC system suction switchover from the condensate storage tank (CST) to the suppression pool when the CST level is low. Until an automatic switchover is implemented, procedures should exist for manual switchover. The originally constructed RCIC system has an automatic switchover capability and is described in section 7.4 of the FSAR. A written procedure for manual switchover is not required, since the RCIC system will be capable of automatic switchover during the unit's initial startup.

II.K.3.24 Confirm Adequacy of Space Cooling For High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems

This item requires the licensee to have RCIC and HPCI systems that are designed to withstand a complete loss of offsite alternating-current power to their support systems, including coolers, for at least 2 hours. Space coolers may be required to maintain the pump room temperatures within allowable limits during long term operation of the RCIC and HPCI systems. NRR has evaluated the licensee's Reactor Building Ventilation System and has documented in section 9.4.2 of the SER that it satisfies the requirements of this item. Each ECCS compartment contains two 100% capacity seismic Category I, Class 1E room coolers which use cooling water supplied from a Category I, Class 1E Emergency Service Water System.

II.K.3.27 Provide Common Reference Level for Vessel Level Instrumentation

This item requires all reactor vessel level instrumentation to be referenced to a common point. Use of a common reference point is intended to help avoid operator confusion. The licensee uses five independent level instrument ranges during the various reactor vessel conditions. The five ranges are: 1) Shutdown water level range, 2) Upset water level range, 3) Narrow water level range, 4) Wide water level range, and 5) Fuel zone water level range. Each of these ranges has a zero reference level at the bottom of the dryer skirt. FSAR section 7.7.1.1.3.1.3 describes the reactor pressure vessel level instruments and has been found to be in compliance with the requirements of this item.

j. (Closed) Item III.A.1.1 Emergency Preparedness Short Term

The licensee's program was evaluated during a recent emergency preparedness appraisal to evaluate its compliance with the requirements of Appendix E to 10 CFR 50 and NUREG 0654. Any identified discrepancies will be tracked separately from this item and therefore this item is considered closed.

k. (Closed) III.A.1.2 Items 1 and 2 Upgrade Emergency Support Facilities

A preliminary inspection of this item was conducted during the emergency preparedness appraisal which reviewed the status of the control room, TSC, EOF and the EFRDS. Final review will be conducted during a future Emergency Response Facility Appraisal which will be documented in a normal inspection report. For administrative purposes, this item is closed.

5. Additional Open Items Reviewed But Not Closed

Also during this reporting period, the following open items were reviewed. The inspectors determined that these items could not be closed for the reasons indicated.

1) Further information required from the licensee:

IEB	80-07	BWR Jet Pump Assembly Failure
IEC	81-03	Inoperable Seismic Monitoring Instrumentation
UNR	50-352/82-05-03	Fail-Safe and Separation Criteria for RPS
UNR	50-352/83-09-01	Use of ASCO Solenoid Valves in High Temperature/Humid Environment
UNR	50-352/83-13-03	System Release to Plant Staff
VIOL	50-352/83-19-03	Recirculation System Pipe Support Design and Installation
UNR	50-352/83-19-09	Preoperational Testing of ESW System
IEC	80-12	Valve-Shaft-to-Actuator Key May Fall Out of Place When Mounted Below Horizontal Axis
VIOL	50-352/84-06-02	Inadequate Corrective Actions for NCR's
VIOL	50-352/84-19-02	Design Change Drawing Controls

2) Licensee Actions Incomplete

IEB	78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures
IEB	80-10	Contamination of Non-Radioactive Systems
IEB	80-12	Decay Heat Removal System Operability
IEB	83-05	Hayward Tyler Pumps

For each of the items listed above, the inspectors informed the licensee of the item status and the reasons why the inspectors found the item not ready for closure.

No violations were identified.

6. Review of Diesel Generator Test Data

The inspector reviewed an internal licensee memorandum, dated 7/24/84, which dealt with the reliability demonstration of the four emergency diesel generators (EDG's). This demonstration was performed during preoperational test 1P24.1 as a result of the licensee's commitment to Position C.2.a(9) of Regulatory Guide (RG) 1.108. The RG Position requires that the preoperational testing program demonstrate adequate system reliability by successfully starting and loading the diesel generators 69 consecutive times on a per-plant basis, with a minimum of 23 consecutive successful tests on each diesel generator unit. According to the memorandum, the licensee concluded that, based on the results of 1P24.1, the Limerick diesel generators were reliable.

The inspector reviewed the data in the memorandum and the notes that described significant test events. The following is a synopsis of the test results in chronological order:

EDG	Consecutive Tests		Significant Event
1. D	5	D.1	On each of the 5 starts, the output frequency was acceptable, but 2.5 Hz higher than the nominal value (60Hz) as a result of a defective component in the electric governor controls. The component (called an EGA) was subsequently replaced.
2. A,B. C,D	49		No significant problems
3. B,C	17	B.1	On the twenty-first start of the B EDG, the output breaker failed to close because of a problem with the breaker spring charging motor.
4. B,C,D	63		No significant events
5. D	0	D.2	On the 32nd start of the D EDG, a contact failed in the engine shutdown relay which, in turn, precluded generator field flashing.

Based on the above data and its interpretation of RG 1.108, the licensee concluded that 134 consecutive successful starts had occurred. Each of the significant events shown above except for D.2 was evaluated and found not to be indicative of a test failure in the context of the RG 1.108 requirements. The inspector disagreed with the licensee's conclusion.

According to RG 1.108, a diesel generator unit is defined to include, among other things, components in the automatic start control system and the diesel generator breaker. The inspector concluded that replacement of the electrical governor component on the D EDG constituted a change to the D diesel generator unit's automatic start controls. Thus no credit could be taken for the 5 starts of the D EDG prior to replacement of the EGA. Further, the inspector determined that the mechanical failure of the B EDG output breaker on the twenty-first start of that unit constituted failure of that unit. Therefore, any successful tests after this failure could not be considered as being consecutive to those which preceded the failure. In summary, the inspector's view was that two series of consecutive successful tests were performed: one of 66 tests and the other of 63 tests. Neither series complied with the 69 consecutive successful start criterion.

The inspector informed the Startup Director and the Vice-President, Engineering and Research of his concerns regarding the conformance of the 1P24.1 results to RG 1.108 Position C.2.a(9) on 7/23 and 7/24 respectively. The inspector further informed the licensee representatives that Region I did not believe further starting of the EDG's was necessary. Rather, Region I would consider the intent of the RG Position to have been met if the licensee would commit to perform additional reliability testing on the EDG output breakers. The acceptability of 1P24.1 is considered unresolved pending further licensee action to demonstrate the reliability of the output breakers. (50-352/84-36-02)

7. New Fuel Inspection and Storage

On July 27, 1984 the licensee received an amendment to License No. SNM-1926 which authorized the licensee to move new fuel bundles from the temporary fuel storage facility to the refueling floor and eventually into the spent fuel pool. Fuel movement began on July 30 and the first fuel bundle was inspected on July 31. An inspection was conducted of the licensee's activities associated with new fuel movement, inspection, channeling and placement into the spent fuel pool on July 30 and July 31, 1984.

The licensee's activities were observed to verify compliance with station procedures, NRC regulations and license conditions. Particular attention was paid to observing compliance with the following fuel handling procedures during performance of the associated activities:

- FH 401 Startup of Refueling Platform
- FH 410 Refueling Platform Shift Checkout
- FH 102 Transfer of New Fuel from Fuel Storage Area to Refuel Floor
- FH 103 Uncrating and Unpacking of New Fuel on the Refueling Floor
- FH 104 Preparation and Shipment of Empty Fuel Boxes
- FH 201 New Fuel Inspection, Channeling and Placement in the Fuel Pool
- FH 210 New Channel Cleaning and Inspection

A copy of each procedure is kept at the reactor engineer's desk on the refueling floor. It was apparent that all personnel involved in these evolutions were familiar with the requirements of these procedures.

The inspector also observed the fuel inspector training provided by a General Electric representative and the initial GE-supervised fuel receipt inspections. Licensee's Quality Control and Quality Assurance personnel were present during the fuel inspection activities and received much of the same training.

The inspector verified that radiological control practices were adequate during fuel inspection and storage activities. This included a review of dosimetry issuance practices as well as direct observation of swipe, direct frisk, and general area radiation surveys. The inspector checked for use of detectors that were in proper calibration and ensured that all applicable pre-use checks had been performed.

After completion of the inspection of a fuel bundle, the fuel channel is placed over the bundle and then the channeled fuel is moved from the fuel inspection stand into the spent fuel pool. The inspector verified that the refueling platform was operated in compliance with the applicable station procedures, and that proper fuel accountability practices were used.

The licensee's fuel inspection, assembly, and storage activities appeared to be well supervised, controlled, and to be in compliance with regulations and procedures. The inspector will continue to inspect these activities. No violations were identified.

8. Unresolved Items

Unresolved items are matters about which more information is necessary to ascertain whether they are violations, deviations, or acceptable items. Unresolved items are discussed in paragraph 6 of this inspection report.

9. Exit Meeting

The NRC resident inspectors discussed the issued and findings in this report throughout the inspection period and at an exit meeting held with Messrs. J. Corcoran and G. Leitch on July 31, 1984.