November 9, 2001

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: LIMERICK GENERATING STATION - NRC INSPECTION REPORT 50-352/01-007, 50-353/01-007

Dear Mr. Kingsley:

On September 28, 2001, the NRC completed a team inspection at your Limerick Generating Station, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on September 28, 2001, with Mr. W. O'Mally, and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety system design and performance capability of the Emergency Service Water (ESW) System and the Automatic Depressurization System (ADS), compliance with the Commission's rules and regulations, and the conditions of your license. Within these areas, the inspection consisted of a selected examination of calculations, drawings, procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the team identified one finding of very low safety significance (Green). This issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating the issue as a Non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at the Limerick Generating Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm.html (The Public Electronic Reading Room).

Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Systems Branch
Division of Reactor Safety

Docket Nos.: 50-352; 50-353
License Nos: NPF-39; NPF-85

Enclosure: NRC Inspection Report No. 50-352/01-007, 50-353/01-007

cc w/encl:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION 1

Docket Nos:	50-352; 50-353
License Nos:	NPF-39, NPF-85
Report No:	50-352/01-007, 50-353/01-007
Licensee:	Exelon Generation Company, LLC
Facility:	Limerick Generating Station, Units 1 & 2
Location:	Evergreen and Sanatoga Roads Sanatoga, PA 19464
Dates:	September 10 - 14, 2001 September 24 - 28, 2001
Inspectors:	J. Yerokun, Team Leader, Division of Reactor Safety (DRS) B. Norris, Senior Reactor Inspector, DRS L. Prividy, Senior Reactor Engineer, DRS A. Della Greca, Senior Reactor Engineer, DRS S. Chaudhary, Senior Reactor Engineer, DRS
Approved by:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000352/01-007, IR 05000353/01-007; on 10/10-10/28/2001; Exelon Generation Company; Limerick Generating Station; Units 1 and 2; Safety System Design and Performance Capability.

This inspection was conducted by five regional inspectors. The inspection identified a Green finding, which was also a non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The team identified a Non-cited violation (NCV) of 10 CFR 50, Appendix B Criterion III, for failure to implement adequate design control measures for the emergency service water (ESW) wetwell screens to verify the adequacy of the design regarding clogging or damage to the screens.

  This finding was determined to be of very low safety significance (Green) by the Significance Determination Process (SDP), Phase 1, because calculations and quarterly pump test results indicated that the screens were not clogged and the ESW system was capable of performing its safety function. (Section 1R21)

Report Details

1. REACTOR SAFETY

Cornerstones: Mitigating Systems and Barrier Integrity

1R19 Post Maintenance Testing (IP 71111.19)

Closed - URI 2000-009-001, RHR, Suppression Pool Spray Function Testing

In NRC Inspection Report 2000-009, the NRC identified that the suppression pool spray safety function of the residual heat removal (RHR) system was not being tested consistent with design requirements. Specifically, in surveillance test, ST-6-051-232-1, "B RHR Pump, Valve and Flow Test," the suppression pool spray mode of RHR was tested by first establishing a RHR loop flow of approximately 5,000 gpm to the suppression pool and then opening the suppression pool spray isolation valve. The inspector found this inappropriate because the test was not performed at the actual design basis RHR loop flow of 10,000 gpm. The issue was left unresolved pending the completion of the licensee's assessment that correlates how the test demonstrated that the suppression pool spray mode design requirements of 500 gpm, with a RHR loop flow of 10,000 gpm, could be met.

During this inspection, the inspectors reviewed the licensing and design bases for the RHR suppression pool spray mode. The review included the technical specifications (TS 3.6.2.2/4.6.2.2 and associated bases); Updated Final Safety Analysis Report (UFSAR) sections 6.2.2, "Containment Heat Removal System," 6.2.1.8.1.1, "Long Term Suppression Pool Temperature Response," and 6.2.1.1.5.1, "Protection Against Bypass Paths"; and standard review plan (SRP) 6.2.1.1.C, Appendix A, "Steam Bypass for Mark I, II, and III containments." The inspector also reviewed the licensee's procedures for initiating and testing the RHR suppression pool spray mode. The review included trip procedure T-225, "Startup and Shutdown of Suppression Pool and Drywell Spray," revision 20; surveillance test ST-6-051-231-1, "A RHR Pump, Valve and Flow Test," revision 42 (ST-6-051-232-1, revision 43 for Unit 1, B loop, and ST-6-051-232-2, revision 31 for Unit 2, B loop)); and field deviation disposition report (FDDR) No. HH2-9066, "Wetwell Spray Flow Testing."

The suppression pool spray design of 500 gpm was based on a total RHR loop flow of 10,000 gpm, with the remaining 9,500 gpm for drywell spray. The design ensured that the containment temperature and pressure limits were not exceeded during a postulated design basis accident condition. The licensee stated that the design was verified during the system pre-operational testing as reflected in FDDR HH2-9066. The TS required surveillance test (ST-6-051-232-1) only demonstrated the functionality of the systems and did not verify the design. The inspector reviewed the licensee's determination and found it acceptable. The inspector reviewed the TS and the surveillance test procedure and concluded that surveillance test met the requirements of the technical specifications. The design requirement for the RHR suppression pool spray mode was verified during the pre-operational testing as documented in FDDR HH2-9066. Therefore, this issue does not result in any finding and is closed.

1R21 Safety System Design and Performance Capability (IP 71111.21)

.1 Emergency Service Water System and Automatic Depressurization System

a. Inspection Scope

This inspection was performed to verify that the design bases have been correctly implemented for the emergency service water (ESW) system and the automatic depressurization system (ADS), such that the systems can be relied upon to meet their functional requirements. The systems were selected because of their significant contribution to core damage frequency as calculated in the Limerick Generating Station (LGS) probabilistic risk assessment (PRA). The ESW system supplies cooling water to essential equipment, such as the emergency diesel generators (EDGs), during a loss of offsite power (LOOP) or loss of coolant accident (LOCA). The ADS is an emergency core cooling system (ECCS) used as a backup to the high pressure coolant injection (HPCI) and/or the reactor core isolation cooling (RCIC) systems to depressurize the reactor so that low pressure coolant injection (LPCI) and/or core spray (CS) can cool the core.

The team reviewed information describing the design and licensing basis functional requirements of the selected systems to verify that: (1) the system design bases were in accordance with the licensing commitments and regulatory requirements; and (2) the design documents, such as drawings and design calculations, were correct. The documents reviewed included the technical specifications (TS), updated final safety analysis report (UFSAR), engineering analysis/calculations, instrument set-point documentation, plant modifications and drawings (piping and instrumentation, isometric, one-line, elementary, electrical, logic & control).

The team reviewed the configuration, operation, testing and maintenance of the systems to determine if they were consistent with their licensing and design bases. The team also reviewed related operating instructions, surveillance and test procedures, normal, abnormal, and emergency operating procedures to determine if they were consistent with design bases and operating assumptions. The review included the system interfaces (instrumentation, controls, and alarms) available to operators to support operator decision making. The team also reviewed the technical specifications required performance data acquired during surveillance testing activities to verify that the results met the acceptance criteria and demonstrated the system's functional capability.

The team assessed the reliability and unavailability performance of the ESW system and ADS by reviewing selected corrective and preventive maintenance work orders (WOs) issued over the past year. The team reviewed post-maintenance testing results for various WOs to verify that they demonstrated the capability of the components to perform their intended safety function.

The team conducted visual inspection and verification for the adequacy of the structural components, such as ESW pump house, equipment and pipe supports, and shock suppressors looking for any evidence of deterioration and/or lack of maintenance.

This inspection also included a review of the implementation of the Agastat relay monitoring program at Limerick Generating Station (LGS). Agastat relays are relied upon to perform critical functions in the reactor protection and safety-related systems of the mitigating system and barrier integrity cornerstones. The team reviewed the following attributes: Component application and performance, service conditions and service life calculations, environmental qualification (EQ), problem identification and resolution, and incorporation of operating experience.

Finally, the team selected a sample of issues associated with the ESW system, ADS and Agastat Relays and reviewed the effectiveness of the licensee's resolution and corrective actions to verify that the licensee was identifying issues at an appropriate threshold and entering them in the corrective action program. The issues selected included those identified by the NRC, the licensee and through the operating experience feedback process.

b. Findings

Emergency Service Water

The team identified a finding concerning the lack of any monitoring or periodic inspections of the stationary screens that are located in the spray pond pumphouse wetwells. The team noted that neither the ESW nor the residual heat removal service water (RHRSW) system design included differential pressure indication or alarm instruments to alert operators for potential blockage of these screens. Design control measures did not verify the adequacy of the design relative to accounting for potential plugging of the wetwell screens. This issue was considered to be of very low safety significance since there was no actual loss of ESW system safety function, and determined to be a non-cited violation (NCV) of 10CFR50, Appendix B, Criterion III, Design Control.

Individual screens with ½" stainless steel mesh are installed in each cubicle for the ESW and RHRSW pumps. Although the screens had not been inspected, the licensee concluded that their conditions were acceptable and that there was no immediate operability concern primarily because the pump quarterly tests did not indicate any degradation of suction performance of the pumps. The licensee confirmed this preliminary conclusion with a calculation completed on September 27, 2001. Using a hydraulic model that simulated different patterns of blockage based on an orifice grid, a weir, and a sluice gate, the licensee calculated the flow through the screens. The orifice grid technique was the most conservative since it resulted in the highest pressure drop. Approximately 70% of the screen open area would have to be clogged to cause a pressure drop of 0.1 psi across the screen. This pressure drop would have a negligible, adverse impact on pump performance. The licensee determined that any current screen blockage was bounded by these results. Therefore, the licensee concluded that the wetwell screens would perform acceptably to support the safety function of the ESW pumps. The licensee indicated that periodic inspections of the stationary screens would be established. The team reviewed the calculation and recent pump quarterly test results, and found the licensee's conclusions reasonable.

In accordance with the NRC Inspection Manual Chapters 0609, “Significance Determination Process,” and 0610, “Power Reactor Inspection Reports,” this issue was determined to be more than minor and worthy of evaluation per the Significance Determination Process (SDP). The lack of adequate design criteria for inspecting the ESW and RHRSW pump stationary screens had a credible impact on safety since a clogged screen could affect the operability of the ESW system and thus affect the mitigating systems cornerstone. When evaluated in accordance with the SDP Phase 1 worksheet, the issue was considered to be of very low safety significance since there was no actual loss of ESW system safety function. Therefore, in accordance with the SDP, Phase 1 Screening, the team determined that this issue was of very low safety significance (Green).

The team concluded that the licensee’s design control measures were inadequate and contrary to the requirements of 10 CFR 50, Appendix B, Criterion III, Design Control, in that measures did not exist for verifying the adequacy of the design. Specifically, design criteria had not been established concerning how much debris could be acceptable to block the screen flow area and not cause ESW pump suction problems. In addition, there was no periodic inspections of the screens. This violation of 10CFR 50, Appendix B, Criterion III, Design Control, is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy issued May 1, 2000 (65FR25368). **(NCV 50-352/2001-007-01, 50-353/2001-007-01).** The licensee incorporated this issue into the corrective action process as CR 00075213, Spray Pond House Screens.

.2 Agastat Relay Failure - High Pressure Coolant Injection

a. Inspection Scope

As a result of the team’s review of Agastat relays, the inspectors identified a potential concern related to the operability of the high pressure coolant injection (HPCI) system. The team reviewed the Technical Specifications, the Licensee Event Report (LER), the surveillance test procedure, the UFSAR and other documents associated with the bases for the original design limits of HPCI injection flow. The review included interviews with the HPCI system engineer and others.

b. Issues and Findings

The failure of an agastat relay and its impact on HPCI injection and anticipated transient without scram (ATWS) design flowrates resulted in Unresolved Item 2001-007-02.

On April 17, 2001, with Unit 2 in a refueling outage, during the performance of ST-6-055-205-2, operators noted that the feedwater injection valve for the HPCI system (HV-055-2F105) did not open due to a failure of an Agastat relay. PEP 10012531 was initiated to address the failure. The relay was replaced and the surveillance test was subsequently re-performed satisfactorily. The surveillance test was to meet the requirements of the Limerick Technical Specification Surveillance Requirement (TSSR 4.5.1.c.1) to verify that each automatic valve in the flow path actuates to its correct position upon performance of a system functional test. The valve had been successfully tested on May 25, 1999. Exelon initially determined that both flow paths (core spray and feedwater lines) were required for HPCI operability and safety function, and submitted Licensee Event Report (LER) 50-353/01-02, in accordance with 10CFR50.73. Subsequently, the LER was retracted based on the determination that the HPCI system would have been able to perform its safety function.

The HPCI system, as described in the Limerick Updated Final Safety Analysis Report (UFSAR), was designed to provide 5,600 gpm cooling water, to ensure that the reactor core was adequately cooled in the event of a small break loss-of-coolant-accident (LOCA) that does not result in rapid depressurization of the reactor vessel, to prevent fuel clad temperatures in excess of the limits (10CFR50.46). The HPCI system injection path is split, 2,500 gpm via the core spray sparger and 3,100 gpm via the feedwater system. The flow split was part of Limerick's mitigation strategy for an anticipated transient without scram. During the surveillance on April 17, 2002, it was identified that the feedwater injection valve would not open, and that all HPCI flow would be via the core spray line only.

The inspectors questioned the licensee's evaluation for past operability. The evaluation referenced in PEP 10012531 did not contain an objective quantitative basis for concluding that the resultant change in the HPCI system flow was acceptable for determining that the system would have been operable and capable of performing its safety functions. Specifically, the GE analysis (GENE-A22-00126-00-01, May 2001) estimated that the flow through the core spray line, although less than the total HPCI design flow but signigicantly greater than the design flow rate through the core spray sparger at rated pressure, would be sufficient for LOCA concerns and would not affect the ATWS mitigation strategy. The GE analysis did not provide the calculations to support that the reduced flow was sufficient to prevent exceeding fuel clad temperatures in the event of a small break LOCA, nor that the increased flow through the core spray line would not create an unanalyzed condition in the event of an ATWS.

The Limerick Design Basis Document (DBD) for the HPCI system stated that the flow split supports the total flow requirement for LOCA and the ATWS requirement imposed on the HPCI system, with a maximum design flow through the core spray system of 3,000 gpm to prevent unacceptable core transients. In addition, the DBD states that the flow split is to mitigate ATWS events, in accordance with NUREG-0460. Section 3.2.1 of the DBD states that the flow rate of 2,000-3,000 gpm through core spray, with the remaining flow through the feedwater system, are "safety-related controlling parameters." The UFSAR, Section 15.8.3.7, Accident Analysis for an ATWS, states that the flow split modification maintains proper HPCI flow mixing with the reactor coolant, and avoids localized fuel channel hydrodynamic effects that might cause local power peaking. Also, UFSAR Table 6.3-1 lists maximum HPCI flow through the core spray sparger (2,890 gpm) as a significant input variable used in the Limerick SAFER/GESTR-LOCA analysis.

In addition, the inspectors identified that the NRC's Safety Evaluation Report (NUREG-0991, August 1983) for the licensee's original submittal of the FSAR stated that the minimum HPCI design flow was 5,600 gpm, and that it discharged into the reactor via a spray sparger. It appeared that the NRC was never formally informed that the HPCI design was changed from that which was originally approved.

This item remains unresolved pending licensee evaluation that the reduced HPCI flow through the core spray system was adequate for LOCA concerns, and the increased flow through the core spray system was not a concern for an ATWS event. **(URI 50-353/2001-07-02)**

4OA6 Meetings, Including Exit

.1 Exit Meetings

The team presented the inspection results to Messrs E. Callan, W. O'Mally, J. Stone, and other members of station management on September 28, 2001. Some proprietary documents were reviewed during the inspection and returned to the licensee. The team verified that the inspection report does not contain proprietary information.

Attachment 1

SUPPLEMENTAL INFORMATION

Key Points of Contact

W. Atsbury, System Manager, ESW
S. Bobyock, Manager, ECCS Systems
C. Brenne, Design Engineer
R. Brown, Emergency Operating Procedures Program Manager
E. Callan, Director, Maintenance
P. Chase, Shift Supervisor
C. Cooney, Manager, Civil/Structural Engineering
M. Crim, Shift Supervisor
K. Dauble, Inservice Test Coordinator
R. Dickinson, Manager, Nuclear Oversight
R. Harding, Regulatory Assurance Engineer
W. Harris, Radiation Protection Manager
J. Krais, Senior Manager, Design Engineering
W. O'Mally, Director, Operations
E. Purdy, System Manager - HPCI/RCIC
T. Ryan, System Manager - ADS
G. Sealy, Manager, Electrical Design
S. Simpson, Manager, Chemistry
J. Stone, Director, Work Management
J. Tucker, Manager, Plant Engineering

Items Opened, Closed, and Discussed

Open and Closed

NCV 2001-007-01	Lack of 10 CFR 50, Appendix B, Criterion III, Design Control Measures for ESW Pump Wetwell Screen. Green. (1R21)

Opened

URI 2001-007-02	Analysis of impact of agastat relay failure on High Pressure Coolant Injection and ATWS design flows. (Section 1R21)

Closed

URI 2000-009-001	RHR Suppression Pool Spray Function Test (1R19)

List of Documents Reviewed

<u>Procedures</u>

E-10/20, Loss of Offsite Power, Rev. 30
ON-113, Loss of RECW, Rev. 20
ON-117, Loss of TECW, Rev. 7
S10.7.A, Abnormal Service Water System Operation, Rev. 27
S11.0.A, Abnormal Operation of ESW System, Rev. 23
S11.1.A, ESW System Startup, Rev. 24
S11.2.A, ESW System Shutdown, Rev. 18
S11.4.B, Drain, Fill, & Vent Unit 2 “B” Loop ESW
S11.4.C, Fill & Vent ESW Piping & Inservice Leak Testing, Rev. 3
S11.6.A, Transfer C & D ESW Pumps to Alternate Power Supply, Rev. 5
S11.7.A, ESW Unitization Operation, Rev. 6
S11.8.A, Alternate Cooling of RECW Heat Exchangers, Rev. 10
S12.7.A, Spray Network to Bypass Transfer, Rev. 12
S12.7.B, Utilization of Cooling Tower or Spray Pond as a Heat Sink for RHRSW/ESW, Rev. 15
S12.7.C, Once Through Operation of ESW/RHRSW, Rev. 12
S12.7.F, Utilizing the ESW/RHRSW Cross-Tie, Rev. 2
S41.0.C, Normal Operation of Suppression Pool Temperature Monitoring System, Rev. 1
S41.0.D, Off-Normal Operation of Suppression Pool Temperature Monitoring System Rev. 12
S41.2.A, Main Steam System Set-Up for Normal Operation, Rev. 2
S41.4.A, Alternate Vessel Letdown Method When RWCU Unavailable, Rev. 3
S41.7.B, SRV’s & Suppression Pool Cooling as an Alternate Shutdown Cooling Method, Rev. 4
S50.1.A, ADS & Main Steam Safety Relief Valve Line-Up, Rev. 6
S53.0.A, Normal Makeup/Response of Low Level in Fuel Storage Pool or Reactor Well, Rev. 18
SE-1, Remote Shutdown, Rev. 50
SE-1-1, Protected Depressurization Control (Long Term Operation), Rev. 12
SE-6, Alternate Remote Shutdown, Rev. 22
ST-1-092-111-1, D11 EDG 4KV SFGD Loss of Power LSF/SAA and Outage Testing. Rev.24
ST-2-011-390-0, ESW/Diesel Generator Heat Transfer Test, Rev 0
ST-2-041-661-1, Safety/Relief Valve Position Indicators Functional Test, Rev. 7
ST-2-050-101-1, Div 1 ADS Logic System Functional/Simulated Auto Actuation, Rev. 3
ST-2-050-600-1, ECCS - ADS Timer; Division 1, Calibration/Functional Test, Rev. 11
ST-2-059-600-1, ECCS-ADS Accumulator Backup Gas Low Pressure Functional Test, Rev. 15
ST-2-088-320-0, Remote Shutdown System ESW & RHRSW Operability Test, Rev. 8
ST-4-041-210-1, Main Steam Relief Valves Test, Rev. 8
ST-4-041-470-1, Cyclic Test of MSSRV Solenoid and Air Operator Assemblies, Rev.2
ST-4-092-912-2, 22 Diesel Generator 18 - Month Inspection, Rev. 2
ST-4-LLR-005-1, “E” Automatic Depressurization System Leak Test, Rev. 4
ST-6-011-203-2, “A” Loop ESW Valve Test, Rev. 14
ST-6-011-231-0, “A” Loop ESW Pump, Valve, & Flow Test, Rev. 46
ST-6-011-232-0, “B” Loop ESW Pump, Valve, & Flow Test, Rev. 50
ST-6-011-364-1, D14 DG ESW IST Valve Indicator Verification Test, Rev. 1
ST-6-011-401-0, Loop “A” ESW Valves Automatic Actuation Test, Rev. 15
ST-6-011-451-0, “A” Loop ESW Lineup Verification, Rev. 39
ST-6-041-201-1, Reactor Vessel Valve Test, Rev. 9
ST-6-055-205-2, HPCI Cold Shutdown Valve Test, Rev. 10
ST-6-107-590-0, Daily Surveillance Log, Common Plant - At All Times, Rev. 57

ST-6-107-591-1, Daily Surveillance Log, OPCON 4,5, Rev. 75
ST-6-107-595-1, Monthly Surveillance Log, OPCON 1,2,3, Rev. 24
RT-2-011-251-0, "ESW Loop A" Flow Balance, Rev. 3
RT-2-011-252-0, "ESW Loop B" Flow Balance, Rev. 4
RT-2-011-391-0, 'A' MCR Chiller Heat Transfer Test, dated 3/6/01, Rev. 2
RT-2-011-392-0, 'B' MCR Chiller Heat Transfer Test, dated 3/13/01, Rev. 1
RT-2-011-392-2, Unit 2C RHR Room Cooler, Air To Water Heat Transfer Test, Rev. 1
RT-2-011-398-2, Unit 2C RHR Motor Oil Cooler Heat Transfer Test, Rev. 4
T-100 to T-117, Trip [emergency operating] Procedures
ARC-MCR-010, Main Control Room Panel 010 Annunciator, Rev. 20
ARC E-10/20, Loss of Offsite Power, Rev. 30
ARC ON-113, Loss of RECW, Rev. 20
ARC ON-117, Loss of TECW, Rev. 7
SAMP-1 & 2, Severe Accident Management Procedures, Rev. 0

Action Requests, Condition Reports and PEPS

A1279192	CR 00061008	PEP I0012531
A1287618	CR 00076445	PEP-I0008508
A1219912	CR 00076456	PEP-I0008973
A1270550	CR 00075213	PEP-I0011188
A1316313	CR 00075094	PEP-I0012312
A1319431	CR 00076400	PEP I0012870
A1281257	CR 00076568	PEP I0011188
A1293845	CR 00076196	PEP I0012308
A1336717	CR 00076525	PEP I0012575
A0899130	CR 00076635	PEP I0012865
A0360314		
A1257898		

Drawings

M-10, Unit 1 & Common P&ID Service Water
M-11, Unit 1, 2 & Common Emergency Service Water
M-0059, Sh. 1-4, Primary Containment Instrument Gas P&ID
E-0015, Sh. 1, Single Line Meter and Relay Diagram, 4 kv Safeguard Power Sys., U1
E-0016, Sh. 1, Single Line Meter and Relay Diagram, 4 kv Safeguard Power Sys., U2
E-0321, Sh. 1-6, Emergency Service Water Pumps
E-0322, Sh. 1,2, D-G ESW Inlet and Outlet MOV's
E-0323, Sh. 1, Turbine Building Cooling Water Heat Exchanger ESW MOV's
E-0324, Sh. 1-3, ESW Discharge to RHRSW MOV's
E-0325, Sh. 1-4, Schematic Diagram Cooling Water Shutoff Valves to SW & ESW-1 & 2 Units
E-0326, Sh. 1, Schematic Diagram ESW Shutoff Valves to RBCW Heat Exchangers, Units 1 & 2
E-0327, Sh. 1-3, Schematic Diagram CR Chiller Cooling Water Shutoff Valves
E-0328, Sh. 1, Schematic Diagram RBCW Heat Exchangers Shutoff Valves to ESW
E-0373, Sh. 1-2, Schematic Diagram RHRSW/ESW to Cooling Tower Shutoff MOVs
E-0374, Sh. 1-2, Schematic Diagram Cooling Tower Return to Spray Pond Shutoff MOVs
E-0375, Sh. 1-3, Schematic Diagram Spray Pond Spray Nozzle Inlet MOVs

E-0376, Sh. 1-2, Schematic Diagram Spray Pond Spray Nozzle Bypass MOVs
E-0377, Sh. 1-2, Schematic Diagram Spray Pond Wetwell Inlet Motor Operated Gates
E-0033, Sh. 1-3, Single Line 125/250 V dc System, Unit 1
E-0034, Sh. 1-3, Single Line 125/250 V dc System, Unit 2
B21-1030-F-001 to 005, Nuclear Boiler System Functional Control Diagram
B21-1060-E-001 to 022, EDS Elementary Diagram, Units 1& 2
FD M-0011, Sh 1-4, Functional Description Emergency Service Water
Civil Drawings, C-1103, 1104, 1107, 1108, 1132, 1138, 1128, and 1151

Design Documents, Calculations & Evaluations

L-S-02, DBD, Emergency Service Water System, Rev. 12
L-S-04, DBD, Residual Heat Removal Service Water
L-S-03, DBD, High Pressure Coolant Injection System, Rev. 17
L-S-31, DBD, Automatic Depressurization System, Rev. 4
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List of Acronyms Used

ADS Automatic Depressurization System
ATWS Anticipated Transient Without Scram
CFR Code of Federal Regulations
CS Core Spray
DBD Design Basis Document
ECCS Emergency Core Cooling System
ESW Essential Service Water
FDDR Field Deviation Disposition Report
GPM Gallons per Minute
HPCI High Pressure Coolant Injection
LER Licensee Event Report
LGS Limerick Generating Station
LOCA Loss of Coolant Accident
LOOP Loss of Off-site Power
LPCI Low pressure Coolant Injection
NPSH Net Positive Suction Head
P&ID Piping and Instrument Diagram
PRA Probabilistic Risk Assessment
RHR Residual Heat Removal
RCIC Reactor Core Isolation Cooling
SDP Significance Determination Process
SRP Standard Review Plan
TDH Total Dynamic Head
TS Technical Specifications
TSSR Technical Specification Surveillance Requirement
UFSAR Updated Final Safety Analysis Report
WO Work Orders