April 30, 2004

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION
NRC INTEGRATED INSPECTION REPORT 05000461/2004002

Dear Mr. Crane:

On March 31, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Clinton Power Station. The enclosed report documents the inspection findings which were discussed on April 7, 2004, with D. Shavey and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, four NRC-identified findings and two self-revealing findings of very low safety significance (Green), five of which were determined to involve violations of NRC requirements, were identified. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these five findings as Non-Cited Violation (NCV)s consistent with Section VI.A of the NRC Enforcement Policy. In addition, one licensee identified violation is discussed in Section 4OA7.

If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the US Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at Clinton Power Station facility.

In accordance with 10 CFR 2.390 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 NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ann Marie Stone, Chief
Branch 3
Division of Reactor Projects

Docket No. 50-461
License No. NPF-62

Enclosure: Inspection Report No. 05000461/2004002
w/Attachment: Supplemental Information

cc w/encl:
Site Vice President - Clinton Power Station
Plant Manager - Clinton Power Station
Regulatory Assurance Manager - Clinton Power Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Vice President - Operations Support
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Enclosure

REGION III

Docket No: License No:	50-461 NPF-62
Report No:	05000461/2004002
Licensee:	AmerGen Energy Company, LLC
Facility:	Clinton Power Station
Location:	Route 54 West Clinton, IL 61727
Dates:	January 1 through March 31, 2004
Inspectors:	B. Dickson, Senior Resident Inspector C. Brown, Resident Inspector R. Alexander, Radiation Specialist M. Holmberg, Senior Reactor Inspector J. Jacobsen, Senior Reactor Inspector P. Lougheed, Senior Reactor Inspector T. Ploski, Senior Emergency Preparedness Inspector T. Tongue, Senior Project Engineer D. Tharp, Reactor Engineer D. Zemel, Illinois Emergency Management Agency
Observers:	C. Roque-Cruz, Nuclear Safety Professional A. Klett, Nuclear Safety Professional C. Acosta, Nuclear Safety Professional
Approved by:	Ann Marie Stone, Chief Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000461/2004002; 01/01/2004 - 03/31/2004; Clinton Power Station; Surveillance Testing, In-Service Inspections, Access Control to Radiologically Significant Areas, and Other Activities.

This report covers a 3-month period of baseline resident inspection and announced baseline inspections on in-service inspection, emergency preparedness, ultimate heat sink inspection and radiation protection. The inspections were conducted by Region III inspectors and resident inspectors. Six Green findings associated with a five non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the sate operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>Inspector-Identified and Self-Revealed Findings</u>

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety-significance and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI. The licensee had replaced shutdown service water (SX) system piping following cavitation induced wall thinning and weld failure leading to a through wall leak in 1999. The corrective actions included periodic non-destructive examination (NDE) monitoring of the pipe-wall for cavitation induced wall-thinning. Following an inquiry by the inspectors about heavy cavitation effects on the piping, the licensee discovered that the NDE monitoring had been performed in the wrong section of the piping. When the correct section was examined, the piping was found below manufacture's minimum allowable wall thickness. The finding affected the cross-cutting area of Human-Performance because the system manager and others had failed to identify that the corrective actions for a previous failed pipe had not been correctly implemented since 1999 and had also subsequently failed to expand the extent of condition to include verifying that all 10 predefined NDE activities established by the 1999 corrective actions were being performed in the correct location immediately downstream of SX system flow orifices.

  The finding was more than minor because it affects the Reactor Safety/Mitigating System Cornerstone and if left uncorrected, it would become a more significant safety concern. The finding was of very low safety-significance because the SX system remained operable, both for function and for seismic considerations. The finding involved the attributes of availability and reliability of the shutdown service water system, internal flooding, and loss of heat sink as well as human performance and could have affected the mitigating systems objective of ensuring the availability of systems that respond to initiating events to prevent undesirable consequences. The licensee entered the event into its corrective action system, performed an operability determination allowing continued use of the pipe, and replaced the piping in March 2004. (Section 1R22)

- Green. A finding of very low safety significance, with an associated Non-Cited Violation, was self-revealed relating to a violation of the requirements of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings. The licensee failed to properly vent the high pressure core spray system before performing an integrated ECCS test resulting in a water-hammer event on the high-pressure core spray system.

  This finding was more than minor because it affected the Mitigating Systems Cornerstone objective of maintaining mitigating systems operable. The finding was of very low safety-significance because a licensee follow-up system investigation, including a complete system walkdown by engineers, revealed that the high pressure core spray system remained operability. This issue was entered into the licensee corrective action program. (Section 1R22)

- Green. A finding of very low safety significance was identified by the inspectors for a violation of the requirements of 10 CFR 50, Appendix B, Criterion III, Design Control. Following the licensee's identification that the operator mounting bolts for several Limitorque SMB-2 actuators did not fit properly, the licensee installed bolts with thread engagement less than the required minimum. This was completed without performing the appropriate level design control review. The minimum thread engagement caused a residual heat removal system Limitorque SMB-2 valve actuator to wobble when operated. This finding affected the cross-cutting area of problem identification and resolution because initially, the licensee did not determine cause or extent of condition of the wobbly actuator.

  This finding was more than minor because it affected the Mitigating Systems Cornerstone objective of maintaining mitigating systems operable. The finding was of very low safety-significance because an evaluation determined that the valve would have performed its safety function when called upon during a design basis seismic event. The finding was entered into the licensee corrective action program and the licensee verified the correct installation of all SMB-2 actuator mounting bolts. (Section 4OA5)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding of very low safety significance associated with inadequate ultrasonic examination procedures used to examine Code welds subject to thermal fatigue.

  This finding was more than minor because it affected the Barrier Integrity Cornerstone objective of maintaining barrier integrity. In this example, the inadequate inservice inspection examination procedures could affect the reactor coolant system barrier integrity in that, if left uncorrected, it could become a more significant safety concern. The inspectors were concerned that if the required examination volumes were not achieved, that the large bore reactor coolant piping would be at an increased risk for failure due to thermal fatigue cracking. Because, there was no evidence of actual flaws, the inspectors concluded that this issue was a finding of very low safety significance. This finding was determined to be a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion IX. (Section 1R08)

- Green. The inspectors identified a finding of very low safety significance associated with an improperly performed a secondary containment draw-down surveillance test. The licensee did not verify the train A standby gas treatment system was capable of drawing a vacuum after an initial test failure. No specific licensee procedure or instruction required by 10 CFR 50 Appendix B was violated; therefore, no violation of regulatory requirements occurred.

  This finding was more than minor because it affected the Barrier Integrity Cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide release caused by accidents or events. The finding was of very low safety-significance because the system was demonstrated operable when properly tested. The licensee entered the event into its corrective action system and performed the test correctly after NRC involvement. (Section 1R19)

Cornerstones: Occupational and Public Radiation Safety (OS)

- Green. A finding of very low safety significance and an associated Non-Cited Violation were identified through a self-revealing event, when on February 6, 2004, an operator working in an area adjacent to the Inclined Fuel Transfer System (IFTS) shield wall in the Fuel Building received an unanticipated electronic dosimetry dose rate alarm. The licensee's subsequent investigation revealed that transfer of spent fuel bundles using the IFTS created a previously unidentified beam of radiation with dose rates in accessible areas in excess of 1000 millirem per hour, and thus the licensee had failed to control the area in accordance with Technical Specifications (i.e., appropriate barricades, postings, and locking mechanisms or flashing lights were not in place).

  The issue was associated with the "Program and Process" attribute of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective in ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material. The issue was more than minor because it involved the occurrence of a potential for unplanned, unintended dose to individuals working in an inadequately controlled high radiation area resulting from conditions contrary to licensee technical specifications and NRC requirements. Based in part on: (1) the dose rates identified in area; (2) the typical spent fuel bundle transit time; and (3) the length of time the operator was in the area, the inspectors determined that there was not an overexposure, nor was there a substantial potential for an overexposure. Therefore, the finding was of very low safety significance. One Non-Cited Violation for the failure to barricade, properly post, and establish a flashing light for the area surrounding the IFTS shield wall in accordance with Technical Specification 5.7.2 was identified.
  (Section 2OS1.1)

B. <u>Licensee-Identified Violations</u>

A violation of very low safety significance, which was identified by the licensee, was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and the licensee's corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The plant entered the inspection period at 70 percent power in response to an electro-hydraulic system leak. Following repairs, the operators raised power to 83 percent (90 percent electrical power) on January 1, 2004. The plant power coasted down to 70 percent (82 percent electrical power) when the operators commenced shutdown for the ninth refueling outage (C1R09). The operators restarted the reactor on February 24, synchronized the generator to the grid on February 27, and reached 92.5 percent reactor power (100 percent electrical power) on February 29, 2004. The operators reduced reactor power to 60 percent for about 14 hours for moisture separator-reheater trouble shooting from March 4 to March 5, 2004. The reactor automatically shutdown from 93 percent power on March 22 in response to a generator trip on high voltage. After repairs to the iso-phase buss ducts, the operators restarted the reactor on March 24, synchronized to the grid on March 25, and raised power to 91.4 percent (full electrical power) on March 26, 2004. The plant was operated at approximately 91.5 percent rated thermal power (maintaining 100 percent electrical output) through the rest of the period except for short periods for control rod pattern adjustments.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors verified that the licensee had completed its seasonal preparations for cold weather in a timely manner before the cold weather actually presented a challenge. The inspectors reviewed the licensee’s completed freezing temperature annual surveillance and verified that it adequately covered risk-significant equipment and ensured that the equipment was in a condition to meet the requirements of Technical Specifications (TSs), the Operations Requirements Manual (ORM), and the Updated Safety Analysis Report (USAR) with respect to protection from low temperatures. The inspectors verified that minor issues identified during the inspection were entered into the licensee’s corrective action system by reviewing the associated condition reports (CRs). Based on their importance for availability of mitigating systems, the inspectors conducted more detailed system reviews and walkdowns for the following two systems:

- Emergency Diesel Generator room and fuel room ventilation
- Reactor core isolation system storage tank and heaters

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04Q)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed partial walkdowns of accessible portions of divisions of risk-significant mitigating systems equipment during times when the divisions were of increased importance due to the redundant divisions or other related equipment being unavailable. The inspectors utilized the valve and electric breaker checklists listed at the end of this report to verify that the components were properly positioned and that support systems were lined up as needed. The inspectors also examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors reviewed outstanding work orders and condition reports (CRs) associated with the divisions to verify that those documents did not reveal issues that could affect division function. The inspectors used the information in the appropriate sections of the Updated Safety Analysis Report (USAR) to determine the functional requirements of the systems. The documents listed at the end of this report were also used by the inspectors to evaluate this area.

The inspectors performed three samples by verifying the alignment of the following systems:

- Residual Heat Removal (RHR) when in shutdown cooling (SDC) Fuel Pool assist mode of operation,

- Division 1 Electrical bus and Division 1 RHR while Division 2 out of service, and

- High Pressure Core Spray during a reactor core isolation cooling work window.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of fire fighting equipment, the control of transient combustibles and ignition sources, and the condition and operating status of installed fire barriers. The inspectors selected fire areas for inspection based on their overall contribution to internal fire risk, as documented in the Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate a plant transient, or their impact on the plant’s ability to respond to a security event. The inspectors used the documents listed at the end of this report to verify that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and that fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program.

The inspectors completed 15 fire protection samples for the following areas:

- Fire Zone CB-1g, control building 781 foot level ,
- Fire Zone C-2, containment outside the drywell,
- Fire Zone A-3f, Division 2 switchgear area,
- Fire Zone A-2o, auxiliary building electrical penetration area,
- Fire Zone A-3e, containment electrical penetration area,
- Fire Zone A-2a, reactor core isolation cooling pump room,
- Fire Zone A-2c, low pressure core spray pump room,
- Fire Zone F-1b, high pressure core spray room,
- Fire Zone F-1p, fuel building 755 foot level,
- Fire Zone A-2b, RHR 'A' pump and heat exchanger rooms,
- Fire Zone A-3a, RHR 'B' pump and heat exchanger rooms,
- Fire Zone A-2n, Division 1 switch gear room,
- Fire Zone D-1, Division 3 HPCS diesel generator oil storage tank room,
- Fire Zone A-2e, MSIV rooms - elevation 737 feet, and
- Fire Areas F-1 (fuel building), A-1, A-2, and A-3 (auxiliary building); fire loading inspection after refueling outage,

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

From March 15 through 19, 2004, specialist inspectors performed the biennial assessment of heat sink performance by reviewing documents associated with the shutdown service water (SX) pump room heat exchangers, the main steam isolation valve leakage control system outboard room cooler, the Division 4 inverter room cooler and the ultimate heat sink. The SX pump room heat exchangers were chosen based on their high risk ranking in the licensee's probabilistic risk assessment. The other two room coolers were chosen due to having inspections performed twice as often as similar equipment. The review of these heat exchangers constituted three samples.

While on site, the inspectors reviewed completed surveillances, associated calculations, instrument calibration records, and maintenance work orders. The inspectors also performed independent calculations to verify that the licensee's activities adequately ensured proper heat transfer. The inspectors reviewed the documentation to confirm that the test or inspection methodology was consistent with accepted industry and scientific practices. The inspectors also reviewed documentation to verify that acceptance criteria were consistent with design basis values as provided in the licensees Generic Letter 89-13 program documentation and that instrument uncertainty was appropriately considered. The inspectors performed a walkdown of the SX system pipe tunnel, the pump rooms and selected room coolers. The inspectors evaluated the ultimate heat sink performance by reviewing pump and valve performance tests and by review of condition reports documenting the results of the ultimate heat sink inspections.

The inspectors also reviewed condition reports concerning heat exchanger, water hammer and radiation alarm issues to verify that the licensee had an appropriate threshold for identifying issues. The inspectors evaluated the effectiveness of the corrective actions for identified issues, including the engineering justification for operability.

The documents that were reviewed are included at the end of the report.

b. Findings

Introduction: An unresolved item was identified with regard to several piping support issues identified during a walkdown of the SX system.

Description: During a March 17, 2004, walkdown of the SX pump rooms and pipe tunnel, the inspectors observed non-conformances with several Division 2 SX piping supports: a rod hanger which was bent, a strut which was out of alignment, and a hanger baseplate which was not flush with the wall.

The licensee had previously identified the bent rod hanger on February 2, 2004. At that time, the licensee presumed the damage to the hanger was caused by a hydraulic transient in an adjacent normal service water pipe, because that pipe had a broken rod hanger. However, the licensee had not taken corrective actions to repair either the broken hanger on the normal service water pipe or the bent rod on the SX pipe hanger. Additionally, the licensee had not taken any actions to ensure that, if the damage was due to a hydraulic transient, that a second transient would not cause further damage in the safety-related piping system.

Following the NRC observations, the licensee took additional corrective actions to examine the welds where the bent rod attached to the embedment plate to ensure that the hanger did not have further damage. The licensee determined that it was unlikely that the damage was caused by a hydraulic transient; however, the apparent cause had not been determined by the end of the inspection. The licensee planned to complete its apparent cause evaluation by the end of April 2004.

In regard to the strut, the licensee performed an immediate operability review and determined that the strut could handle its design basis loading. The licensee planned to also complete the apparent cause evaluation for this issue by the end of April 2004.

In regard to the hanger baseplate, the licensee determined that the gap exceeded the maximum allowable gap in licensee procedure 8199.01, "Concrete Expansion Anchor Work." Based on this determination, the licensee immediately shimmed the baseplate to bring it into conformance with the standard. The licensee also performed additional walkdown of the Division 2 SX system and identified four additional baseplates which did not meet the procedural requirements. One of these additional baseplates was also immediately shimmed, while the other three were determined to not require shimming. The licensee's initial operability determination concluded that the hangers for all five cases were likely acceptable in the as-found condition, based on the actual maximum loading on the hanger versus the design value.

The licensee believed that the cause of the baseplate gaps probably stemmed from initial construction. Based on this conclusion, the inspectors questioned the acceptability of other baseplates on the other division of SX as well as other safety-related systems. The licensee stated that the extent of the condition was limited to small-bore piping (piping under 2 inches, nominal) because the large bore piping was routinely inspected, because the large bore SX piping in the pump rooms and pipe tunnel were welded to embedded plates, and because the small bore baseplates used a standard installation which assumed a bounding loading. However, the inspectors could not determine if the licensee's assumptions would apply to other systems.

As stated above, the bent support and mis-aligned strut had not been fixed by the end of the inspection. Neither had the licensee determined the cause of the degraded supports. Therefore, the inspectors were unable to conclude that the SX system would not be subject to further damage which might render the system unable to perform its safety function. Further, the inspectors were unable to conclude that the licensee's assumptions regarding the hanger baseplates were bounding, given the licensee's postulation that the gaps may have existed since initial construction.

This item is being held as an unresolved item pending additional information from the licensee determining the cause of the degraded supports and baseplates, the extent of condition, and the severity and impact on affected piping and piping supports. The licensee documented these issues in CRs 199858, 209240, 211005, and 211161.
(URI 05000461/2004002-01)

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's Inservice Inspection Program for monitoring degradation of the reactor coolant system boundary and risk significant piping system boundaries, based on review of records of nondestructive examinations.

On February 10 - 11, 2004, the inspectors observed licensee vendor personnel:

- Performing ultrasonic (UT) examination of a 10-inch residual heat removal system weld 1-RH-14-24 inside the main steam tunnel.

- Performing magnetic particle examination of a pipe to penetration weld (1-RH-14-30) (Code Class 2 piping) inside the main steam tunnel.

- Performing remote visual VT-one examination of reactor vessel surveillance capsule holder welds at the 177 and 183 degree azimuth locations inside the reactor vessel. This inspection was conducted by vendor personnel with pole mounted cameras and was observed by the inspectors from the work platform positioned above the vessel cavity.

The inspectors reviewed the UT examination records for two reactor vessel head meridian welds (Class 1 Welds).

The inspectors completed the observations and record reviews discussed above to confirm that the ASME Code Section V and Section XI requirements were met. The inspectors concluded that this review counted as two inspection samples as described in Section 71111.08-5 of inspection procedure 71111.08, "Inservice Inspection Activities."

The inspectors also reviewed ultrasonic examination reports for the reactor vessel recirculation inlet N2 nozzle weld completed on April 10, 2002, and the reactor vessel head dollar plate weld completed on April 15, 2002. During these examinations, the licensee identified relevant indications. The inspectors conducted a review of these examinations to confirm that the licensee had correctly evaluated and dispositioned these indications in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. The inspectors concluded that this review counted as one inspection sample as described in Section 71111.08-5 of inspection procedure 71111.08, "Inservice Inspection Activities."

The inspectors reviewed the pressure boundary weld records for welds 3604-01 and 3604-02. These welds were fabricated during replacement of the reactor core isolation cooling steam admission valve 1E51-F045 (Class 2 component). The inspectors conducted this review to confirm for these welds, that the welding acceptance and preservice examinations (e.g., radiography, pressure testing, weld metal tensile tests and bend tests) were performed in accordance with ASME Code, Section III, Section V, Section IX, and Section XI. The inspectors concluded that this review counted as one inspection sample as described in Section 71111.08-5 of Inspection Procedure 71111.08, "Inservice Inspection Activities."

The inspectors also reviewed the licensee's records associated with two ASME Section XI Code repair/replacement activities for valves 1E51-F045 and 1E51-F041C.

Specifically, the inspectors reviewed the Code required documentation including pressure tests completed for the reactor core isolation cooling steam admission valve 1E51-F045 and modifications to 1E51-F041C which is a residual heat removal loop C isolation check valve. The inspectors conducted this review to confirm that the ASME Code Section III, Section V, and Section XI requirements were met for these activities. The inspectors concluded that this review counted as one inspection sample as described in Section 71111.08-5 of Inspection Procedure 71111.08, "Inservice Inspection Activities."

b. Findings

Inadequate Ultrasonic Examination Procedures for Welds Subject To Thermal Fatigue

Introduction: The inspectors identified a finding of very low safety significance and an associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion IX, related to inadequate ultrasonic examination procedures used to examine Code welds subject to thermal fatigue.

Description: The inspectors identified that procedure GE-PDI-UT-1, "PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds," Revision 1, and Procedure GE-UT-209, "Procedure for Automated Ultrasonic Examination of Dissimilar Metal Welds, and Nozzle to Safe End Welds," did not contain the extent of scanning (e.g., volume to be examined) for welds subject to thermal fatigue. The licensee had used these procedures during the last refueling outage to examine Code welds subject to thermal fatigue.

In April of 2002, the licensee examined a reactor vessel nozzle to shell weld (N4B-W-1) and a reducer to pipe feedwater weld (1-FW-1-1-2) using procedures GE-UT-209 and GE-PDI-UT-1 respectively. These welds were identified as subject to thermal fatigue (Examination Category R1.11) in the licensee's Inservice Examination Program which was based upon the Electric Power Research Institute's Revised Risk-Informed Inservice Inspection Procedure (TR-112657). For these components, the licensee should have examined to 1/4-inch beyond the counterbore at the inside of these components to identify any fatigue cracking which would have initiated from the counterbore in accordance with Figure 4.2 of TR-112657. However, for these welds, the examination records did not contain explicit documentation of the extent of scanning achieved and the inspectors could not independently confirm that the minimum extent of scanning was completed. Therefore, the inspectors were concerned that the licensee may not have identified fatigue cracking initiated from the piping counterbore region, if the required expanded weld examination volume was not completed. The licensee documented this concern in Issue Reports 00200725 and 00200733.

Analysis: The inspectors determined that the inadequate procedure was a performance deficiency warranting a significance determination. The inspectors reviewed this finding against the guidance contained in Appendix B, "Issue Dispositioning Screening," of IMC 0612, "Power Reactor Inspection Reports." In particular, the inspectors compared this finding to the findings identified in Appendix E, "Examples of Minor Issues," of IMC 0612 to determine whether the finding was minor. Following that review, the inspectors concluded that none of the examples listed in Appendix E accurately represented this example. As a result, the inspectors compared this performance deficiency to the minor questions contained in Section 3, “Minor Questions,” to Appendix B of IMC 0612. The inspectors concluded that the finding was greater than minor because the finding was associated with the Barrier Integrity (BI) Cornerstone attribute and affected the BI objective of maintaining barrier integrity. In this example, the inadequate inservice inspection examination procedures could affect the reactor coolant system barrier integrity in that, if left uncorrected, it could become a more significant safety concern. The inspectors were concerned that if the required examination volumes were not achieved, that the large bore reactor coolant piping would be at an increased risk for failure due to thermal fatigue cracking. For the feedwater welds N4B-W-1 and 1-FW-1-1-2, the licensee’s examination records did not contain explicit documentation to confirm the extent of scanning achieved and therefore, the inspectors considered that this issue was of more than minor significance.

The inspectors evaluated this finding using Manual Chapter 0609, “Significance Determination Process,” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening associated with the Barrier Integrity Cornerstone. Based upon this screening, the inspectors determined that a risk evaluation using Phase 2 worksheets was required. Because there was no evidence of actual flaws for the pressure boundary weld examinations reviewed, the inspectors did not assign any increase in frequency to the large break loss of coolant accident sequences in the Phase 2 worksheets. Without any increase in frequency of large break loss of coolant accident sequences, the inspector determined that the core damage frequency associated with this finding remained unchanged. Therefore, the inspectors concluded that this issue was a finding of very low safety significance (Green).

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion IX, required, in part, that measures shall be established to assure that special processes including nondestructive testing are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes and standards. The ASME Code 1995 Edition, 1996 Addenda of Section XI, Appendix VIII, Article VIII-2100, “Procedure Requirements,” states, in part, that the examination procedure shall specify essential variables including the extent of the scanning. The extent of scanning for welds subject to thermal fatigue was defined in Figure 4.2 of TR-112657, “Revised Risk-Informed Inservice Inspection Procedure.” Contrary to these requirements, procedure GE-PDI-UT-1, “PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds,” Revision 1, and procedure GE-UT-209, “Procedure for Automated Ultrasonic Examination of Dissimilar Metal Welds, and Nozzle to Safe End Welds,” did not contain the extent of scanning (e.g., volume to be examined) for welds subject to thermal fatigue. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (Condition Reports 00200725 and 00200733), it is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000461/2004002-02)

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

The inspectors reviewed licensed-operator requalification training to evaluate operator performance in mitigating the consequences of a simulated event, particularly in the areas of human performance. The inspectors evaluated operator performance attributes which included communication clarity and formality, timely performance of appropriate operator actions, appropriate alarm response, proper procedure use and adherence, and senior reactor operator oversight and command and control.

Crew performance in these areas was compared to licensee management expectations and guidelines as presented in the following documents:

- Operator exam per ESG-LOR-70, "Loss of 1A - Rod Drift - ATWS," Revision 00, including the trainers critique, the operators' self critique, and simulator fidelity.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's maintenance efforts in implementing the maintenance rule (MR) requirements, including a review of scoping, goal-setting, performance monitoring, short-term and long-term corrective actions, and current equipment performance problems. These systems were selected based on their designation as risk significant under the MR, or their being in the increased monitoring (MR category (a) (1)) group. The three inspection samples were completed by review of the following systems:

- Low pressure core spray room,

- Safety Relief Valve Replacement, and

- Shutdown Service Water check valves.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors observed the licensee's risk assessment processes and considerations used to plan and schedule maintenance activities on safety-related structures, systems, and components particularly to ensure that maintenance risk and emergent work contingencies had been identified and resolved. The inspectors completed eight samples by assessing the effectiveness of risk management activities for the following work activities or work weeks:

- CPS 3503.01, Class 1E swing battery charger 1DC11E feed to safety related bus inspection and testing,

- WO# 656030, Self test system automatic depressurization system (channel 4),

- Replacement of automatic depressurization system #4 card WO#656030,

- RHR ‘A’ Heat Exchanger inspection and maintenance,

- Division-1 bus outage and Integrated Test,

- Verification of risk profile and awareness of Division 2 RHR/Reactor Core Isolation Cooling (RCIC) isolation capabilities during Div. 1 RHR/RCIC isolation L.S.T. leak detection,

- CPS 9080.21 Division 1 integrated test, and

- CPS 9054.01C002, Reactor Core isolation Cooling High Pressure Operability checks,

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14)

a. Inspection Scope

The inspectors reviewed personnel performance during planned and unplanned plant evolutions and selected licensee event reports focusing on those involving personnel response to non-routine conditions. The review was performed to ascertain that operator’s responses were in accordance with the required procedures. In particular, the inspectors reviewed personnel performance during the following two plant events:

- Bottom head drain line temperature rate-of-change event on shutdown for refueling outage 1CR09 and

- Plant automatic shutdown following turbine generator trip on over voltage on March 22, 2004.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following operability determinations and evaluations affecting mitigating systems to determine whether operability was properly justified and the component or system remained available such that no unrecognized risk increase had occurred.

The inspectors completed five samples of the operability determinations and evaluations.

- Division-1 Emergency Diesel Generator with mis-matched engine output,
- High Pressure Core Spray (HPCS) water hammer evaluation,
- OE 197883, “Division 3 Degraded Voltage Time Delay,” Revision 0,
- Operability determination associated with CR 200534 HPCS water hammer during Division 3 integrated testing, and
- High/Low SSW Pump Room 1B set point issue (50 degrees) 50524F, WO 653843.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

Cumulative Effects of Operator Workarounds

a. Inspection Scope

The inspectors completed a cumulative effect review of all open operator workarounds on the reliability, availability and potential for misoperating a system. Additionally, the inspectors evaluated the effect of the operator workaround on the operator’s ability to implement abnormal or emergency operating procedures.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed one permanent plant modification to verify that the instructions were consistent with applicable design modification documents and that the modifications did not adversely impact system operability or availability. The inspectors interviewed operations, engineering and maintenance personnel as appropriate and reviewed the design modification documents and the 10 CFR 50 Part 50.59 evaluations against the applicable portions of the USAR. The documents listed at the end of this report were also used by the inspectors to evaluate this area.

The inspectors also verified that permanent plant modifications performed during increased risk-significant configurations do not place the plant in an unsafe condition. The inspectors reviewed the following design change package:

- Engineering Change 339004 Rev. 1, Change Boron 10 enrichment in standby liquid control tank, WO#489967.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the post maintenance testing activities associated with maintenance or modification of important mitigating, barrier integrity, and support systems that were identified as risk significant in the licensee's risk analysis. The inspectors reviewed these activities to verify that the post maintenance testing was performed adequately, demonstrated that the maintenance was successful, and that operability was restored. During this inspection activity, the inspectors interviewed maintenance and engineering department personnel and reviewed the completed post maintenance testing documentation. The inspectors used the appropriate sections of the TS and USAR, as well as the documents listed at the end of this report, to evaluate this area.

The inspectors completed nine samples by observing and evaluating the following:

- CPS 9065.02 Secondary containment integrity using standby gas treatment Train 'A', WO 452030-01,

- Division-1 emergency diesel generator (EDG) testing after adjustments under WO 00656431,

- Division 2 battery cell replacement,

- Division 2 Emergency Diesel Generator actuator replacement,

- Main steam isolation valve isolation system response time test WO 529365,

- CPS 9015.02 Standby liquid control inject operability follow maintenance,

- WO 531246 -Replacement/Removal of snubber 1RH090555 RHR common shutdown cooling suction line,

- Reactor Core Isolation Cooling and high pressure testing as post maintenance testing for pump rebuild, and

- WO 669291 - Backfill of HPCS suppression pool level detector 1E22-N655C

b. Findings

Containment Draw Down Test

Introduction: The inspectors identified a finding of very low safety significance associated with an improperly performed secondary containment integrity testing.

Description: On January 8, 2004, the licensee performed Clinton Power Station (CPS) 9065.02 “Secondary Containment Integrity” test to fulfill an 18-month surveillance requirement (TS 3.6.4.1.4). This surveillance test acceptance criteria required the licensee to verify each standby gas treatment subsystem will draw down the secondary containment to greater that 0.25 inch of vacuum water gauge within a specified time period. The TS also required that the standby gas treatment subsystem used to perform this surveillance test requirements be staggered.

During the surveillance test, the licensee started the A Train of standby gas treatment system and determined that the train could not meet the surveillance requirement test acceptance criteria. The licensee declared secondary containment inoperable, and entered TS limiting condition for operation action statement 3.6.4.1A.1. The licensee performed various corrective actions such as repairing several secondary containment door seals. Following these repairs, the licensee performed CPS 9065.02 a second time, this time using the B Train of standby gas treatment system. The B Train standby gas treatment system was successful in meeting CPS 9065.02 acceptance criteria. The licensee declared secondary containment operable and exited TS action statement.

On January 9, 2004, following a review of the surveillance test information, the inspectors concluded that by not testing the A Train of standby gas treatment system which failed the original surveillance test, the licensee did not meet the intent of Regulatory Guide 1.52, which was to have two redundant divisions of standby gas treatment system, each capable of quickly drawing down the secondary containment environment to a negative 0.25-inch vacuum on system start-up.

The licensee agreed with the inspector’s conclusions. On January 10, 2004, the licensee re-performed CPS 9506.02 using the A Train of standby gas treatment. During this test the licensee observed that the flow rate during the draw down test exceeded the maximum flow rate of 4400 scfm shown on Figure 1 of CPS 9506.02. Since the flow rate was beyond the maximum allowable flow rate of 4400 scfm shown in the surveillance test procedure, the acceptability of the observed draw down time could not be determined. Excessive airflow through the standby gas treatment system could reduce the removal efficiencies committed to in the USAR, possibly to the extent where offsite and control room dose limits as specified in 10 CFR 100 are challenged. The licensee performed an operability evaluation (OE 194488) and concluded that the test result was acceptable and that there was no impact to operability based on engineering judgement. The system was declared operable.

Analysis: The inspectors determined that failing to test the A Train standby gas treatment system after an initial test failure was a performance deficiency warranting a significance evaluation. The inspectors reviewed this finding against the guidance contained in Appendix B, “Issue Dispositioning Screening,” of IMC 0612, “Power Reactor Inspection Reports.” In particular, the inspectors compared this finding to the findings identified in Appendix E, “Examples of Minor Issues,” of IMC 0612 to determine whether the finding was minor. Following that review, the inspectors concluded that none of the examples listed in Appendix E accurately represented this example. As a result, the inspectors evaluated this performance deficiency using the minor questions contained in Section 3, “Minor Questions,” of Appendix B of IMC 0612. The inspectors concluded that the finding was greater than minor because the finding was associated with the Barrier Integrity (BI) Cornerstone attribute and affected the BI objective of providing reasonable assurance that physical design barriers protect the public from radionuclide release caused by accidents or events. In this example, the failure to re-perform the containment draw down test could have led to a division of standby gas treatment to not be able to quickly draw down the secondary containment environment to a negative 0.25-inch vacuum on system start-up.

The inspectors evaluated this finding using Manual Chapter 0609, “Significance Determination Process,” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening associated with the Barrier Integrity Cornerstone. The inspectors answered “no” to all three questions, under Containment Barriers. Therefore, the inspectors concluded that this issue was a finding of very low safety significance (Green).

Enforcement: This issue was the result of the licensee failing to meet the intent of Regulatory Guide 1.52. No specific licensee procedure or instruction required by 10 CFR 50 Appendix B was violated; therefore, no violation of regulatory requirements occurred. This issue was considered a finding of very low safety significance (FIN 05000461/2004002-03). The licensee entered the event into its corrective action system as CR 194448.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated the licensee’s conduct of refueling outage activities to assess the licensee’s control of plant configuration and management of shutdown risk. The inspectors reviewed configuration management to verify that the licensee maintained defense-in-depth commensurate with the shutdown risk plan; reviewed major outage work activities to ensure that correct system lineups were maintained for key mitigating systems; and observed refueling activities to verify that fuel handling operations were performed in accordance with the TS and approved procedures. These activities represent one inspection sample.

The inspectors performed the following activities daily, during the outage:

- attended control room operator and outage management turnover meetings to verify that the current shutdown risk status was well understood and communicated;
- performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk;
- observed the operability of reactor coolant system instrumentation and compared channels and divisions against one another;
- performed walkdowns of the auxiliary and containment buildings to observe ongoing work activities; and
- reviewed selected issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance.

Additionally, the inspectors performed the following specific activities:

- observed Reactor Core Isolation Cooling suppression pool level transmitter replacement PMID (0015897202),
- verified out of service T/O DC-A02,
- observed source range monitor functional check CPS 90000.01D001, and compared channels against one another,
- observed reactor disassembly,
- verified Residual Heat Removal A alignment CPS 9027.01C001 RSP operability,
- verified outage configuration management and shutdown risk assessment,
- observed first enforcer platform lift, move and set,
- verified preparations to move the B recirculation pump motor,
- verified clearance orders for Division 1 bus outage and RHR ‘A’ Heat Exchanger outage,
- ensured correlation of level instrumentation during the cooldown,

- verified electrical line ups during various buss outages and after the integrated testing.

- verified lineups and status of the decay heat removal systems,

- verified water level after cavity flood up.

- verified readiness for fuel moves and observed fuel moves from the control room and in containment, and

- monitored reactor shutdown including turbine trip due to vibrations and preparations for forced cooldown.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed selected surveillance testing and/or reviewed test data to verify that the equipment tested using the surveillance procedures met the TS, the ORM, the USAR, and licensee procedural requirements, and demonstrated that the equipment was capable of performing its intended safety functions. The activities were selected based on their importance in verifying mitigating systems capability and barrier integrity. The inspectors used the documents listed at the end of this report to verify that the testing met the frequency requirements; that the tests were conducted in accordance with the procedures, including establishing the proper plant conditions and prerequisites; that the test acceptance criteria were met; and that the results of the tests were properly reviewed and recorded. In addition, the inspectors interviewed operations, maintenance and engineering department personnel regarding the tests and test results.

The inspectors completed 15 samples by evaluating the following surveillance tests:

- CPS 9052.01 LPCS/RHR A Pump & LPCS/RHR A Water Leg Pump Operability,

- CPS 9080.01 Diesel Generator 1A Operability - Manual and Quick Start Operability monthly surveillance test,

- CPS 9065.02 Secondary Containment Integrity test,

- CPS 9067.03 Standby Gas Treatment System Train A test conducted on January 8, 2004,
- CPS 3506.01 Diesel Generator and support systems,
- CPS 9609.01 Shutdown Service Water Operability test, Division 2 WO 654249,
- CPS 9333.40 Division-III, 4.16 kV bus under-voltage relay (degraded voltage) final functional test,
- Division 1 EDG run to support data taking per CPS 2700.23 Diesel Generator Diagnostic Testing,
- Control room ventilation (VC) ‘B’ outage and operability testing,
- CPS 9051.01 HPCS Pump and Water Leg Pump Operability,
- CPS 9080.21 Division-1 Emergency Diesel Generator integrated testing,
- CPS 9080.23 Diesel generator 1C-ECCS integrated performance test,
- CPS 9091.02 Refuel Bridge Crane/Hoist operability,
- CPS 9080.25 DG 1B test mode override load reject operability, and idle speed override, and
- Shutdown Service Water Cavitation-Induced Wall Thinning.

b. Findings

(1) High Pressure Core Spray Pressure Transient

Introduction: A Green finding with an associated Non-Cited Violation (NCV) was self-revealed when an inadequate procedure resulted in a hydraulic pressure transient (water hammer) during the Division 3 Integrated ECCS surveillance test.

Description: On February 9, 2004, the licensee performed CPS 9080.23 “Division 3 Integrated ECCS”. During the performance of CPS 9080.23, the high pressure core spray system was started. Immediately following the system start, licensee staff noted a very loud noise. The inspectors, who were in containment, also heard a very loud sound. During the follow-up investigation of this event, the licensee identified that an unusually high pressure transient occurred immediately following the start of the high pressure core spray system. Peak system pressure of 1687 psig was recorded. The licensee review of the pressure transient indicated that the pressure was much higher than normally seen during the system’s quarterly surveillance test.

The licensee’s follow-up investigation of this issue determined that the HPCS system was not properly vented before the system was started. Specifically, the licensee identified that Section 8.1.3 of CPS 9080.23, which required the licensee to verify HPCS system was filled and vented, did not contain steps to verify that the piping section downstream of the high pressure core spray injection valve, 1E22-F004, and upstream of the manual isolation valve, 1E22-F036, was vented.

During this surveillance test, the 1E22-F036 was closed. Prior to the performance of CPS 9080.23, the licensee performed a local leak rate test on the HPCS system discharge check valve using air. The licensee suspected that air was introduced in the discharge piping during the local leak test and migrated into the section of piping between the 1E22-F004 and 1E22-F036 valves. The licensee determined that the loud noise originated from the abrupt cycling of the discharge check valve 1E22-F005.

As part of the licensee investigation of this event, the licensee staff performed an operability evaluation which concluded that the system's ability to perform it safety function was not affected. Additionally, the licensee's engineering staff walked down all the piping in the high pressure core spray system. No evidence of water hammer damage or piping movement was noted. The licensee documented the results of their evaluation in Engineering Change Request, (ECR) 363628 and ECR evaluation 347393.

Analysis: The deficiency associated with this event was an inadequate procedure, which led to the high pressure core spray system experiencing a water hammer event. Licensee procedures did not have steps to ensure the system was properly vented and filled before it was started. The inspectors used IMC 0612, Appendix B, to disposition this issue and determined that it was more than minor because the finding affected the Reactor Safety/Mitigating System objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors entered the significance determination process using Manual Chapter 0609, Appendix A, "Significance Determination For Reactor Inspection Findings For At-Power Situations." Using the SDP Worksheet 1, the inspectors answered "yes" to question 1 in the Phase I analysis under the mitigating system cornerstone which resulted in the finding screening out as Green. Based on this conclusion, this finding was determined to be of very low safety significance (Green).

Enforcement: 10 CFR 50, Appendix B, Criterion V, requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawing of a type appropriate to the circumstances. Contrary to this, on February 9, 2004, during the performance of the Division 3 integrated test the high pressure coolant injection system experienced an hammer event due to an inadequate procedure. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000461/2003009-04). This issue was entered into the licensee corrective action program in Condition Reports 201268, 200632, and 200534.

(2) Cavitation-Induced Shutdown Service Water Pipe-Wall Thinning

Introduction: The inspectors identified a Non-Cited Violation (NCV) of Criterion XVI of 10 CFR 50, Appendix B, having very low safety significance (Green) for inadequate corrective actions to address cavitation induced pipe-wall and flange thinning in the shutdown service water (SX) system piping downstream from the Division 3 EDG flow orifice. This issue was considered to be NRC-identified because it was identified after the inspectors questioned the status of piping experiencing heavy cavitation leading to the licensee discovering that previous pipe-wall thickness examinations (in 2002) had not been accomplished in the area of most concern for cavitation damage.

Description: On August 27, 2003, the inspectors walked down the Division 3 EDG room after the completion of scheduled EDG operations. The inspectors noted that the shutdown service water (SX) piping leading from the EDG was emitting loud noises and oscillating such that the piping supports and hangers were moving; all symptoms of cavitation in the SX water. The inspectors asked operators in the area if the noise and shaking piping were usual after an EDG run. The operators investigated and then initiated CR 173409, “Possible Cavitation of Div 3 DG HX [heat exchanger] Outlet Valve 1SX006C.” The licensee initiated actions to evaluate the noise (cavitation) and vibration during the cooldown of the Div III Diesel, with a due date of December 5, 2003.

In late October, the inspectors were informed that the piping in question had been replaced in 1999 (corrective actions from Condition Report 1-99-04-303) and a program of regular ultrasonic (UT) wall thickness measurements were being performed (additional corrective actions) due to the previous cavitation induced wall thinning event. The system manager stated that the UT inspection had been performed in 2002 (a 3-year periodicity) with satisfactory results. The inspectors requested a copy of the results from the 2002 UT inspection. A few days later, the system manager informed the inspectors that the UT measurements had not been taken in the flange to pipe weld area immediately downstream of the orifice (the area of concern). It was determined that the first examination points on the inspection grid were 6 inches downstream from the flow orifice. On November 5, 2003, the system manager initiated CR 184946, “Insufficient NDE [non-destructive examination] of Pipe Section,” to document this issue and to initiate additional UT examination.

As an immediate corrective action, the licensee performed a UT at the desired location and determined that the actual measured wall thickness of the SX piping was as low as 0.120 inches. Nominal pipe-wall thickness for Schedule 40, 8-inch piping is 0.322 inches and the manufacturer’s recommended minimum wall thickness is 0.282 inches (87.5 percent of nominal). On November 14, 2003, the system manager initiated CR 186659, “SX System Piping Wall Thinning,” to document that the results of the additional UT examination had revealed a highly localized area of wall thinning on the bottom of the pipe just downstream of flow orifice 1SX14MC (discharge piping from Division 3 EDG). The licensee completed an immediate operability determination and evaluation (OE186659 -08) which determined that the piping remained functional, that structural integrity was not impaired and that ASME Code Requirements were being met as long as the minimum pipe wall at that area was at least 0.08 inches. The licensee calculated that the pipe wall was eroding at a rate of 0.0037 inches per month (based on the time in service from 1999) and that it would take about 10 months (August 2004) to reach the calculated minimum code wall thickness of 0.08 inches. Based on the OE, the licensee declared the piping operable until it could be replaced in the next applicable work window. The licensee replaced the flange, 6 feet of piping, and the downstream elbow in March 2004.

The inspectors noted that CR 1-99-04-303 documented pipe-wall thinning that had required this same pipe section to be replaced in 1999. At that time, the thinnest measurement found at the toe of the pipe to flange weld, on the bottom of the pipe immediately next to the orifice, was 0.210 inches. However, the UT examination (work document F00212 dated April 23, 1999) attached to CR 1-99-04-303 documented that the flange to pipe weld had cracked at the 5 o'clock position causing a small through wall leak. The licensee had discovered the cavitation induced erosion when attempting to weld repair this presumed "pinhole" leak. In CR 1-99-04-303, the licensee determined that the premature thinning and failure of the 1SX04AC pipe was caused by insufficient design provisions to control a combination of corrosion mechanisms and cavitation erosion from flow through the orifice into the low pressure region downstream of the orifice. The licensee established corrective action 3 which required initiating regular inspections of all areas immediately downstream of orifices in the SX system for pipe-wall thinning due to cavitation. In 1999, the licensee initiated actions to perform UT inspections on the 18 orifices that were not already being monitored.

During this current inspection period, the inspectors reviewed the 10 predefined work activities, generated from CR 1-99-04-303 and noted that the locations to perform the UTs was general (i.e. "downstream of the orifices). The inspectors also determined that the licensee's extent of condition did not include a review of the exact areas under going UT examination for the other orifices.

<u>Analysis</u>: The inspectors determined that failing to ensure that the corrective actions from CR 1-99-04-303 were correctly implemented, to prevent the recurrence of significant pipe-wall thinning from cavitation induced erosion, was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening. The finding was more than minor because it affected the Reactor Safety/Mitigating System Cornerstone and if left uncorrected, it would become a more significant safety concern. The finding involved the attributes of availability and reliability of the shutdown service water system, internal flooding, and loss of heat sink as well as human performance and could have affected the mitigating systems objective of ensuring the availability of systems that respond to initiating events to prevent undesirable consequences. Unmonitored cavitation induced pipe-wall thinning could cause the loss of a division of the shutdown service water system as well as flooding and loss of heat sink. The finding also affected the cross-cutting area of Human-Performance because the system manager and others failed to identify that the corrective actions for a previous failed pipe had not been correctly implemented and subsequently failed to expand the extent of condition for the issue. The inspectors entered the significance determination process using Manual Chapter 0609, Appendix A, "Significance Determination For Reactor Inspection Findings For At-Power Situations." Using the SDP Worksheet 1, the inspectors answered "no" to all five questions in the Phase I analysis under the mitigating system cornerstone which resulted in the finding screening out as Green. Based on this conclusion, this finding was determined to be of very low safety significance (Green).

<u>Enforcement</u>: 10 CFR 50, Appendix B, Criterion XVI, requires, in part, that corrective actions be taken to preclude repetition of significant conditions adverse to quality. Condition Report 1-99-04-303, "During weld repairs per F00212 there was an area of

pipe found to be below minimum wall thickness," Revision 0, identified significant cavitation induced pipe-wall thinning and a cracked weld leading to a through wall leak on the Division 3 SX system. Corrective Action 3 to CR 1-99-04-303 required the establishment of 10 predefined work activities to periodically perform non-destructive examination of all previously unmonitored SX piping sections downstream of flow orifices for cavitation induced pipe-wall thinning. Contrary to this, between October 20, 1999, and November 14, 2003, the licensee failed to implement corrective actions to prevent repetition of a significant condition adverse to quality. Specifically, the licensee did not ensure that predefined non-destructive examinations, a corrective actions for a previous significant condition adverse to quality, were performed at the area immediately downstream of flow orifice 1SX14MC. As a result, significant unmonitored cavitation-induced wall-thinning had occurred. The licensee entered the issue into the corrective action program as CR 184946 and CR 185659. The licensee completed an operability determination and found the piping to be operable and then replaced the piping in March 2004. Because this violation was of very low safety significance and it was entered into the corrective action program, this violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000461/2004002-05)

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed temporary plant modifications to verify that the instructions were consistent with applicable design modification documents and that the modifications did not adversely impact system operability or availability. The inspectors interviewed operations, engineering and maintenance personnel as appropriate and reviewed the design modification documents and the 10 CFR 50.59 evaluations against the applicable portions of the USAR. The documents listed at the end of this report were also used by the inspectors to evaluate this area.

The inspectors reviewed the issues that the licensee entered into its corrective action program to verify that identified temporary modification problems were being entered into the program with the appropriate characterization and significance. The inspectors also reviewed the licensee's corrective actions for temporary modification related issues documented in selected condition reports. The condition reports are specified in the List of Documents Reviewed.

The inspectors reviewed and evaluated the following temporary plant modifications on risk-significant equipment:

- Temporary Mod Review (EC 342243 000) "RHR B Heat Exchanger Temporary Conductivity Monitor."

- Temporary Mod Review (EC 347123 000) "Add Temporary Vibration Sensors to Turbine Control Valves to Monitor EHC Line Vibration During Normal Operation Until C1R10."

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector reviewed Revision 5 of the Clinton Power Station Annex to the Exelon Standardized Emergency Plan to determine if changes identified in this annex revision reduced the Plan's effectiveness, pending on-site inspection of the implementation of these changes.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed licensee controls and surveys for selected radiation areas, high radiation areas and airborne radioactivity areas in the following four radiologically significant work areas within the plant and reviewed work packages which included associated licensee controls and surveys for these areas to determine if radiological controls (including surveys, postings and barricades) were acceptable:

- Containment Building,
- Drywell,
- Fuel Building, and
- Radwaste Building.

The inspectors reviewed the radiation work permits (RWP) and work packages used to control work in these four areas and other high radiation work areas to identify the work control instructions and control barriers that had been specified. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. Workers were interviewed to verify that they were aware of the actions required when their electronic dosimeters noticeably malfunctioned or alarmed.

The inspectors walked down these four areas to verify that the prescribed RWPs, procedures, and engineering controls were in place, that licensee surveys and postings were complete and accurate, and that air samplers (if necessary) were properly located.

Though there were no highly activated and/or contaminated materials (non-fuel) stored within spent fuel pool at the time of the inspection, the inspectors reviewed the licensee's physical and programmatic controls for such items should the licensee have the need to store those materials in the spent fuel pool.

These reviews represented four inspection samples.

b. Findings

Introduction: One self-revealed Green finding and an associated Non-Cited Violation (NCV), were identified when, on February 6, 2004, an operator received an unanticipated electronic dosimetry (ED) dose rate alarm while working in the vicinity of the Inclined Fuel Transfer System (IFTS) Tube shield wall on the 755 foot elevation of the Fuel Building. The licensee's subsequent investigation revealed that transfer of spent fuel bundles using the IFTS created previously unidentified dose rates in accessible areas in excess of 1000 millirem per hour, and thus the licensee had failed to control the area in accordance with TSs (i.e., appropriate barricades, postings, and locking mechanisms or flashing lights were not in place).

Description: On February 6, 2004, at approximately 8:45 p.m., an operator was in the process of hanging a routine outage clearance for a transmitter in the area of the shield block for the Inclined Fuel Transfer Tube on the 755 foot elevation of the Fuel Building. At the time, the area where the operator worked was controlled as a Radiation Area based on the results of historical surveys performed in the area. During his activities, the operator received an ED dose rate alarm in excess of the dose rate alarm set point of 100 millirem per hour established by the RWP that governed the work. The operator recognized the alarm, exited the radiologically controlled area, and reported to the Radiation Protection (RP) Control Point. The RP staff immediately investigated the unanticipated dose rate alarm and determined that the maximum dose rate recorded by the ED was 234 millirem per hour, and the operator's recorded accumulated dose was 0.6 millirem for the brief entry.

Based on the initial investigation into the ED dose rate alarm and current outage activities, RP and Operations Supervision determined that a spent fuel bundle was transferred via the IFTS from the 828 foot elevation pool to the 755 foot elevation pool at approximately the time of the operator's ED alarm. At that point all fuel movements were stopped, additional radiological controls (i.e., barricades, postings, and flashing lights) were established around the area, and the licensee initiated a prompt investigation.

Licensee investigatory surveys as the next fuel bundle passed through the area identified the source of radiation to be an approximately 1-inch diameter beam exiting the wall at the interface between the west side of the IFTS shield wall and the Reactor Containment wall, approximately 12 feet above the 755 foot elevation floor level. RP personnel identified that the source of radiation streamed downward at an approximately 30 degree angle. Dose rate measurements along the 30 degree angle path of streaming radiation were as follows:

- 150 rem per hour at the exit point from the wall,
- 100 rem per hour at 12 inches from the exit point,
- 5 - 7 rem per hour, 6 feet above the 755 foot elevation floor level (approximately head level), and
- 0.6 rem per hour, 4.5 feet above the 755 foot elevation floor level and 20 feet from the IFTS shield wall/Containment wall interface (approximately chest level).

The licensee's prompt investigation also evaluated whether there were any operational changes which could account for the unanticipated dose rates in the Fuel Building. In particular, the licensee evaluated the core exposure rates for the spent fuel bundles during C1R09, and IFTS water levels in the system during the fuel offloads. However, spent fuel bundle core exposure rates were consistent with previous bundles during the offload, and the IFTS water filling system was operable and water levels were consistent with previous bundle transfers. Additionally, the licensee did not identify any anomalous gaps in the shield wall/containment wall interface, and the caulk in the shield wall/Containment wall interface was observed as only having minor superficial irregularities.

The licensee had performed contact gradient surveys of the areas around the IFTS shield wall during the station's first three refueling outages in the 1989 - 1993 time frame, but no streaming radiation was identified by those surveys. The results of these historical surveys were used to establish the radiological controls in this area during refueling outages through February 2004 (during C1R09). However, the licensee noted in the prompt investigation that the gradient surveys would likely have been conducted while standing on the floor level, and thus a beam exiting the shield at 12 feet above the floor level could have been missed.

Finally, based on the radiation survey results, and the location of and time spent in the area by the operator, the licensee preliminarily calculated the dose to the operator at 2 to 3 millirem to the highest exposed portion of the whole body, though the ED recorded only 0.6 millirem for the entry.

<u>Analysis</u>: The inspectors determined that the licensee failed to meet the requirements of TS 5.7.2, in that: (1) the requisite high radiation area (HRA) controls were not in place, nor anticipated, when the operator received the ED dose rate alarm; and (2)

surveys of reasonably accessible areas above the floor level (6 to 7 feet) indicated dose rates in excess of 1000 millirem per hour. The performance deficiency is associated with the "Program and Process" (RP Controls) attribute of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective in ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material. Also, the issue involved the occurrence of a potential for unplanned, unintended dose to individuals working in an inadequately controlled HRA resulting from conditions contrary to licensee TSs and NRC requirements. Therefore, the issue was more than minor and represents a finding which was evaluated using the significance determination process (SDP) for the Occupational Radiation Safety Cornerstone.

The inspectors determined utilizing Manual Chapter 0609, Appendix C, "Occupational Radiation Safety SDP," that the finding did not involve ALARA/work controls. Further, based on: (1) the dose rates identified in accessible areas near the IFTS shield wall; (2) the size and directional nature of the radiation beam; (3) the typical spent fuel bundle transit time through the area (up to 45 seconds); and (4) the length of time the operator was in the area, the inspectors determined that there was not an overexposure, nor was there a substantial potential for an overexposure. Additionally, the licensee's ability to assess dose was not compromised. Consequently, the inspectors concluded that the SDP assessment for this finding was of very low safety significance (Green).

Enforcement: Technical Specification 5.7.2 requires, in part, that "individual high radiation areas in which an individual could receive a deep dose equivalent greater than or equal to 1000 millirem in 1 hour (at 30 centimeters), accessible to personnel . . . where no lockable enclosure can be reasonably constructed . . . shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device." Contrary to the above, on February 6, 2004, it was identified that the licensee failed to barricade, properly post, and establish a flashing light for the area surrounding the IFTS shield wall on the 755 foot elevation of the Fuel Building, when spent fuel transfers via the IFTS created dose rates in accessible areas of 5000 - 7000 millirem per hour. However, because the licensee documented this issue in its corrective action program (CR 200209), took immediate corrective actions to appropriately control the area and initiated a prompt investigation, and the violation is of very low safety significance, it is being treated as a Non-Cited Violation (NCV 05000461/2004002-06).

.2 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed10 corrective action reports related to access controls written during the most recent C1R09 refueling outage, including reports on high radiation area radiological incidents, as available. Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- Initial problem identification, characterization, and tracking,
- Disposition of operability/reportability issues,

- Evaluation of safety significance/risk and priority for resolution,
- Identification of repetitive problems,
- Identification of contributing causes,
- Identification and implementation of effective corrective actions,
- Resolution of Non-Cited Violations tracked in the corrective action system, and
- Implementation/consideration of risk significant operational experience feedback.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.3 Job-In-Progress Reviews

a. Inspection Scope

The inspectors observed the following three jobs that were being performed in radiation areas, airborne radioactivity areas, or high radiation areas for observation of work activities that presented the greatest radiological risk to workers:

- Loop 'B' Reactor Recirculation System Work (specifically the pump internals lift and removal) [RWP No. 10002826],
- Suppression Pool Diving for Debris Removal [RWP No. 10002862], and
- Undervessel Low Range Power Monitor (LPRM) Removal Activities [RWP No. 10002839].

The inspectors reviewed radiological job requirements for these three activities, including RWP and work procedure requirements, and attended ALARA pre-job briefings.

Job performance was observed with respect to these requirements to verify that radiological conditions in the work areas were adequately communicated to workers through pre-job briefings and postings. The inspectors also verified the adequacy of radiological controls (including required radiation, contamination, and airborne surveys); radiation protection job coverage (including audio/visual surveillance for remote job coverage); and contamination controls.

Radiological work in high radiation work areas having significant dose rate gradients was reviewed to evaluate the application of dosimetry to effectively monitor exposure to personnel and to verify that licensee controls were adequate. In particular, the suppression pool diving activities and the retrieval of an LPRM detector (which had fallen to the floor of the undervessel) involved areas where the dose rate gradients were severe which increased the necessity of providing multiple dosimeters and/or enhanced job controls.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.4 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated radiation protection work requirements and evaluated whether workers were aware of the significant radiological conditions in their workplace, the RWP controls and limits in place, and that their performance accounted for the level of radiological hazards present.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.5 Radiation Protection Technician Proficiency

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation protection technician performance with respect to radiation protection work requirements and evaluated whether they were aware of the radiological conditions in their workplace, the RWP controls and limits in place, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

2OS2 As Low As Is Reasonably Achievable Planning And Controls (ALARA) (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the C1R09 refueling outage work scheduled during the inspection period and associated work activity exposure estimates for the following six work activities which were likely to result in the highest personnel collective exposures:

- Drywell Steam Relief Valve (SRV) Work [RWP No. 10002827],
- Loop ‘B’ Reactor Recirculation System Work [RWP No. 10002826],
- Undervessel LPRM Removal Activities [RWP No. 10002839],
- Source Range Monitor (SRM)/Intermediate Range Monitor (IRM) Detector Repair, Replacement, and Maintenance Activities [RWP No. 10002838],
- Drywell Scaffold Construction/Removal Activities [RWP No. 10002842], and
- Drywell Reactor Water Cleanup System Work [RWP No. 10002841].

Additionally, the inspectors reviewed licensee procedures associated with maintaining occupational exposures ALARA and processes used to estimate and track work activity specific exposures.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors evaluated the licensee’s list of work activities ranked by estimated exposure that were in progress and reviewed the following five work activities of highest exposure significance:

- Drywell SRV Work,
- Loop ‘B’ Reactor Recirculation System Work,
- Undervessel LPRM Removal Activities,
- SRM/IRM Detector Repair, Replacement, and Maintenance Activities, and
- Drywell Scaffold Construction/Removal Activities.

For these five activities, the inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements in order to verify that the licensee had established procedures, and engineering and work controls that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The interfaces between radiation protection, operations, maintenance, planning, scheduling and engineering groups were evaluated by the inspectors to identify interface problems or missing program elements. Finally, the inspectors evaluated the integration radiological job planning activities (pre-job ALARA reviews) into work procedure and RWP documents.

These reviews represented four inspection samples.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the licensee's process for adjusting exposure estimates or pre-planning work, when unexpected changes in scope, emergent work or higher than anticipated radiation levels were encountered. This review included a determination if adjustments to estimated exposures (intended dose) were based on sound radiation protection and ALARA principles, rather than adjustments to account for failures to adequately control the work. The frequency of these adjustments was reviewed to evaluate the adequacy of the original ALARA planning process. In particular, the inspectors reviewed the licensee's Work-In-Progress reviews conducted for those C1R09 activities in which significantly higher than anticipated dose rates (i.e., increased source term) were encountered (Drywell SRV and Reactor Water Cleanup System work activities).

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors observed the three activities identified in Section 2OS1.3 that were being performed in radiation areas, airborne radioactivity areas, or high radiation areas for observation of work activities that presented the greatest radiological risk to workers. The licensee's use of ALARA controls for these work activities was evaluated using the following:

- The licensee's use of engineering controls to achieve dose reductions was evaluated to verify that procedures and controls were consistent with the licensee's ALARA reviews, that sufficient shielding of radiation sources was provided for and that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding.

- Job sites were observed to determine if workers were utilizing the low dose waiting areas and were effective in maintaining their doses ALARA by moving to the low dose waiting area when subjected to temporary work delays.

These reviews represented two inspection samples.

b. <u>Findings</u>

No findings of significance were identified.

.5 <u>Radiation Worker Performance</u>

a. <u>Inspection Scope</u>

Radiation worker and RP technician performance was observed during work activities performed in radiological areas that presented the greatest radiological risk to workers. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope and tools to be used, by utilizing ALARA low dose waiting areas, and that work activity controls were being complied with. Also, radiation worker performance was observed to determine whether individual training/skill level was sufficient with respect to the radiological hazards and the work involved.

These reviews represented one inspection sample.

b. <u>Findings</u>

No findings of significance were identified.

4 OTHER ACTIVITIES (OA)

4OA1 <u>Performance Indicator Verification</u> (71151)

To perform a periodic review of performance indicator (PI) data to determine its accuracy and completeness.

Cornerstones: Initiating Events

.1 Reactor Safety Strategic Area

a. Inspection Scope

The inspectors sampled the licensee's submittals for performance indicators (PIs) for the specified period through December 31, 2003. The inspectors used PI definitions and guidance contained in Revision 2 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline" to verify the accuracy of the PI data.

The inspectors performed three samples by reviewing the following:

- Scrams with loss of normal heat removal.
- Unplanned Scrams per 7,000 Critical hours.
- Unplanned power changes per 7,000 critical hours.

b. Issues and Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the licensee's corrective action system as a result of inspectors' observations are included in the list of documents reviewed which are attached to this report.

b. Findings

No finding of significance were identified.

.2 Review of Prompt Investigations for Trends, Rigor, and Common-Cause Attributes (Annual Sample)

Introduction

The number of plant issues receiving a prompt investigation had increased from two in the last 6 months of 2003 to seven in the first 2 months of 2004. The inspectors selected eight condition reports with an approved prompt investigation from July 2003 through February 2004 for an annual sample review of the licensee's problem identification and resolution program.

a. Inspection Scope

The inspectors reviewed the CRs and the associated prompt investigations for any discernible trends, common-cause attributes, and lack of rigor. The CRs were as follows:

- CR 168629, Abnormal RPV Level/Loss of Feedwater at Power; dated July 22, 2003, and the prompt investigation dated July 25, 2003.
- CR 188839, Reactor Scram, dated December 2, 2003, and the prompt investigation dated December 5, 2003.
- CR 194301, Empty New Fuel Box Departs W/O Proper Documentation, dated January 1, 2004, and the prompt investigation dated January 14, 2004.
- CR 195216, Sediment Pond PH Above NPDES Allowable Value, dated January 14, 2004, and the prompt investigation dated January 16, 2004.
- CR 199043, Incorrect Fuel Bundle Grappled and Lifted, dated February 2, 2004, and the prompt investigation dated February 4, 2004.
- CR 200209, ED Dose Rate Alarm Received During Fuel Transfer, dated February 7, 2004, and the prompt investigation dated February 10, 2004.
- CR 201760, Finger Injury to Mechanical Maintenance Mechanic, dated February 14, 2004, and the prompt investigation dated February 17, 2004.
- CR 202233, Near Miss 1E12F009 Valve Started to Operate, dated February 17, 2004, and the prompt investigation dated February 19, 2004.

The inspectors reviewed the above CRs and the associated prompt investigations to verify that the licensee's identification of the problems were complete, accurate, and timely, and that the consideration of extent of condition review, generic implications, common cause, and previous occurrences was adequate. The inspectors also considered the licensee's evaluation and disposition of performance issues, evaluation and disposition of operability issues, and application of risk insights for prioritization of issues.

b. Issues

No findings of significance were identified. The inspectors did not identify a discernible trend or common-cause for the issues documented by the selected CRs. Two involved communication errors -- one by licensee staff and the other by an outside truck driver. One was caused by equipment failure and another by using the wrong equipment for a task. The remaining four CRs had elements of human performance but no repetitive error modes. The inspectors reviewed the proposed corrective actions for the selected CRs to ensure that generic implications were addressed and that the corrective actions were appropriately focused on correcting the identified problems.

4OA3 Event Follow-up (71153)

.1 On January 8, 2004, the inspectors monitored the licensee's immediate corrective actions in response to a failure of the secondary containment integrity test. The inspectors observed the licensee's initial assessment, LCO entry, equipment status adjustments, successful follow-up test and timely LCO exit. Long term corrective actions will be reviewed under the routine inspection program. No immediate findings of significance were identified.

.2 On March 22, 2004, the inspectors monitored the licensee's immediate corrective actions in response to an automatic plant shutdown from 93 percent power. The inspectors observed the licensee's immediate actions and performance under the emergency operating procedures (EOPs), including notification to the NRC. The inspectors also reviewed equipment status and response to the automatic shutdown until the plant was stable. The initial evaluation determined that the automatic shutdown was in response to a main turbine trip on generator over-voltage. The licensee immediately formed a troubleshooting team to investigate the cause of the over-voltage condition on the main generator. Long term corrective actions will be reviewed under the routine inspection program. No immediate findings of significance were identified.

4OA4 Cross-Cutting Aspects of Findings

.1 A finding described in Section 1R22 had as a primary cause, a human performance deficiency in that the system manager and others failed to identify that the corrective actions for a previous failed pipe had not been correctly implemented and subsequently failed to expand the extent of condition for the issue.

.2 A finding described in Section 4OA5 had as a primary cause, a problem and identification deficiency in that the licensee did not initially evaluate the extent of condition or cause of a wobbly valve actuator.

4OA5 Other Activities

.1 Spent Fuel Material Control and Accounting At Nuclear Power Plants (TI 2515/154)

a. Inspection Scope

The inspectors interviewed the station and Exelon corporate special nuclear material custodians. The inspectors reviewed licensee procedures regarding the movement and accountability of special nuclear material. The inspectors also reviewed a sample of recent inventories of nuclear fuel and special nuclear material. Clinton Station has not had a fuel element failure and has never disassembled an used-fuel bundle. Documents reviewed as part of this TI are listed in the Attachment. This TI was not a part of the baseline inspection program and was therefore not considered a sample. The TI is considered complete.

b. Findings

No finding of significance were identified.

.2 (Closed) Unresolved Item 05000461/2003004-01: Appropriateness of Residual Heat Removal Heat Exchanger Capacity Calculation Model

In response to this unresolved item, the inspectors reviewed the licensee's methodology for determining the residual heat exchanger heat transfer capability. The inspectors confirmed that the licensee was appropriately modeling the heat exchanger in accordance with the tubular exchanger manufacturers association (TEMA) requirements as a TEMA-E shell and tube heat exchanger. The licensee used an appropriate correction factor to the log mean temperature difference to account for the fact that the shell and tube fluids were not totally in a counterflow configuration. The inspectors verified that use of this factor was in accordance with standard heat transfer methodologies. The inspectors further confirmed that the licensee was using industry standard computer software to analyze the results of the performance testing. The inspectors had previously independently confirmed that the software provided reasonable results in regard to heat exchanger performance. The inspectors determined that the licensee was doing performance testing at an appropriate frequency to identify any degradation of heat transfer capability. No violations of NRC requirements were identified and this item is closed.

.3 (Closed) Unresolved Item 05000461/2003004-02: Licensee's Rationale for Concluding No Residual Heat Removal Heat Exchanger Tube Leakage

In response to this unresolved item, the inspectors discussed the results of the eddy current test on the Division 1 residual heat removal heat exchanger with the licensee. The inspectors also reviewed the results of the weekly grab samples from the Division 2 SX radiation monitor and reviewed the actions taken in response to several condition reports on radiation monitor alarms. The inspectors determined that the Division 1 residual heat removal heat exchanger had very little degradation, with only four tubes requiring preventive plugging. In regard to the Division 2 residual heat removal heat exchanger, the inspectors determined that the grab samples were taken at least weekly as well as following any alarms from the radiation monitor. The inspectors determined that the results of the grab sample spectroscopy analysis showed no radioactive isotopes present in the SX. Therefore, the inspectors concluded that the licensee's program was adequate to ensure the integrity of the heat exchangers and that no leaks existed on the residual heat removal heat exchangers. No violations of NRC requirements were identified and this item is closed.

.4 <u>(Closed) URI 05000461/2003005-01:</u> Thread Engagement of Limitorque Actuator Mounting Bolts less than vendor recommendations.

<u>Introduction:</u> The inspectors identified 1 Non-Cited Violation (NCV) of very low safety significance (Green) as a result of the licensee failing to evaluate a potential configuration change in accordance with the licensee's established configuration control process following the installation of modified mounting bolts in Limitorque SMB-2 actuators.

<u>Description:</u> As discussed in IR 05000461/2003005, on September 23, 2003, while performing preventive maintenance activities on the residual heat removal heat exchanger bypass valve (1E12-F048B), the licensee observed that the motor-operated valve actuator wobbled as maintenance workers manually stroked the valve. The actuator used in this application was a Limitorque SMB-2.

In IR 05000461/2003005, the inspectors considered the licensee failure to properly evaluate a potential configuration change a violation of Appendix B, 10 CFR Part 50, Criteria III, which stated that design changes, including field changes, shall be subjected to design control measures commensurate with those applied to the original design. However, during that time the inspectors were unable to assess the finding in accordance with IMC 0609,"Significance Determination Process" due a number of open issues. Resolution of following issues were needed to the complete the significance determination process:

- Extent of condition for this issue.

- The adequacy of having a minimum of 5/8 inch thread engagement.

- The effect of an non-conservative assumption made in CPS calculation IP-CL018.

- Past operability of wobbly residual heat removal heat exchanger bypass valve (1E12-F048B).

<u>Extent of Condition</u>:

The licensee performed a detailed extent of review investigation. Based on this review, the licensee concluded that there was no evidence to suggest any additional action are required.

<u>Adequacy of in 5/8 Inch Thread Engagement</u>

As stated in IR 05000461/2003005, by the end of the inspection period, the licensee completed inspection of five of the seven SMB-2 Limitorque actuators used in safety-related systems. These valves were located in both divisions of the residual heat removal system. The licensee noted that each of the five actuators had operator mounting bolts which did not meet the previously established minimum thread engagement. The shortest thread engagement length was on the residual heat removal heat exchanger outlet valve (5/8 inch thread engagement). The licensee evaluation of this concluded that with a minimum of 5/8 inch thread engagement, SMB-2 Limitorque actuators would continue to perform their safety function during a design basis earthquake.

<u>The Effect of an Non-conservative Assumption Made in CPS Calculation IP-CL-018</u>;

As discussed in IR 05000461/2003005, the inspectors questioned engineering assumptions made by the licensee in calculation IP-CL-018. For example, the licensee assumed that the bolts effective grip length was equal to the length of the bolts thread engagement. The inspectors concluded that this assumption resulted in a higher than actual joint stiffness factor. This higher than actual joint stiffness factor would result in a non-conservative outcome when calculating the remaining joint force as thread engagement is decreased from the 1 inch minimum requirement. The licensee completed a re-assessment of the accuracy of the calculation and determined that the stress on the casting threads decreased (calculation was conservative), but the error impacted the joint compression force. In this calculation, there was sufficient preload so the mating members will remain in compression. Therefore, there was no adverse impact to the calculation results.

<u>Past operability of wobbly residual heat removal heat exchanger bypass valve (1E12-F048B)</u>

In IR 05000461/2003005, the inspectors questioned the past operability of residual heat removal heat exchanger bypass valve 1E12-F048B. For approximately 2 years, the actuator remained in a condition such that, under design basis seismic loading, the valve's functionality was questionable. The licensee completed an evaluation to determine the wobbly valve's past operability. In this evaluation, the licensee concluded that the valve would have performed its safety function when called upon during a design basis seismic event.

<u>Analysis:</u> The inspectors considered the licensee's failure to properly evaluate a potential configuration change a performance deficiency. The inspectors used IMC 0612, Appendix B, to disposition this issue and determined that it was more than minor because the finding was associated with the Mitigating System crosscutting attribute of Equipment Performance and affected the Mitigating System objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, failure to properly analyze a configuration change could lead to degradation in the valve's ability to perform appropriately during a design basis accident. The inspectors entered the significance determination process using Manual Chapter 0609, Appendix A, "Significance Determination For Reactor Inspection Findings For At-Power Situations." Using the SDP Worksheet 1, the inspectors answered "yes" to question 1 in the Phase I analysis under the mitigating system cornerstone which resulted in the finding screening out as

Green. Based on this conclusion, this finding was determined to be of very low safety significance (Green).

This finding also affected the cross-cutting area of problem identification and resolution because as described in IR 05000461/2003005, the licensee did not initially determine the programmatic cause or extent of condition of the wobbly actuator.

Enforcement: 10 CFR Part 50, Appendix B, Criteria III, states that design changes, including field changes, shall be subjected to design control measures commensurate with those applied to the original design. Exelon procedure CC-AA-10 “Configuration Control Process Description” stated that a “design input” comprises the criteria, parameters, bases, assumptions and other design requirements upon which detailed final design is based. Appendix I of CPS 8451.04 required a minimum thread engagement length of 1-inch mounting bolts for the Limitorque SMB-2 valve actuator in question. Contrary to the above, about 2 years ago, following the licensee identification that the operator mounting bolts for several Limitorque SMB-2 did not fit properly, the licensee installed bolts with thread engagement less than the required minimum. This was completed without performing the appropriate level design control review. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000461/2004002-07). This issue was entered into the licensee corrective action program in Condition Reports 201268, 200632, and 200534.

4OA6 Meetings

.1 Exit Meetings

The inspectors presented the inspection results to Mr. D. Shavey and other members of licensee management at the conclusion of the inspection on April 7, 2004. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Inservice Inspection (IP 7111108) with Mr. R. Bement and other members of your staff on February 13, 2004.

- Occupational Radiation Safety ALARA and Access Control Programs inspection with Mr. R. Bement on February 13, 2004.

- Emergency Preparedness inspection with Mr. S. McCain on March 5, 2004.

4OA7 Licensee-Identified Violations

The following violation of very-low significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600, for being disposittioned as NCVs.

Cornerstone: Mitigating Systems

Technical Specification Surveillance test Requirement 3.3.6.1.5, T1.f, requires that the main steam line area turbine building temperature monitors be channel calibration checked every 18 months plus a 25 percent grace period per CPS 9432.29, "MSL Area Turbine Building Temperature 1E31-N559A (B,C,D), 1E31-N560A (B,C,D), 1E31-N561A (B,C,D), 1E31-N562A (B,C,D), 1E31-N563A (B,C,D) Channel Calibration. As described in CR 208801, 1E31-N561C was not calibrated between July 31, 2000, and December 5, 2002, a period of 844 days. The maximum allowable period was 684 days. When 1E31-N561C was tested in December 2002, the as-found and the as-left data was satisfactory.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
R. Bement, Site Vice President
M. McDowell, Plant Manager
D. Anthony, NDE Engineer
M. Baig, ISI Engineer
K. Baker, Design Engineering Senior Manager
B. Bunte, Engineering Programs Manager
J. Cunningham, Work Management Director
H. Do, Exelon ISI Engineer
R. Davis, Radiation Protection Director
R. Frantz, Regulatory Assurance Representative
M. Friedman, Emergency Preparedness Manager
M. Hiter, Access Control Supervisor
W. Iliff, Regulatory Assurance Director
J. Madden, Nuclear Oversight Manager
S. McCain, Corporate Emergency Preparedness Manager
G. Mosely, Performance Engineering
S. O'Reilly, 89-13 Program Coordinator
J. Peterson, Regulatory Assurance Representative
R. Schmidt, Maintenance Manager
D. Schavey, Operations Director
J. Sears, Chemistry Manager
T. Shortell, Training Manager
J. Wade, Radiation Protection - ALARA
C. Williamson, Security Manager
J. Williams, Site Engineering Director
R. Zacholski, Shift Operations Superintendent

NRC
J. Wigginton, Senior Health Physicist, NRR

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000461/2004002-01	URI	Pipe Support Issues Identified on Division 2 SX Piping

Opened and Closed

05000461/2004002-02	NCV	Inadequate Ultrasonic Examination Procedures for Welds Subject to Thermal Fatigue

05000461/2004002-03	FIN	Containment Draw Down Post Maintenance Testing
05000461/2004002-04	NCV	Emergency Core Cooling System Water Hammer
05000461/2004002-05	NCV	Ineffective Corrective Action Pipe Wall Thinning
05000461/2004002-06	NCV	Failure to Establish Appropriate Radiological Controls for a TS High Radiation Area
05000451/2004002-07	NCV	Design Control of Motor Operated Valve Mounting Bolts

Closed

05000461/2003004-01	URI	Appropriateness of RHR Heat Exchanger Capacity Calculation Model
05000461/2003004-02	URI	Licensee's Rationale For Concluding No RHR Heat Exchanger Tube Leakage

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any pat of it, unless this is stated in the body of the inspection report.

<u>1R04</u> <u>Equipment Alignments</u>

CPS 3312.01, Fuel Pool Cooling assistance
CPS 9035.01, RHR/LPCS valve operability shutdown

<u>1R05</u> <u>Fire Protection</u>

Updated Safety Analysis Report
Fire Protection Evaluation Report
Fire Protection Safe Shutdown Analysis

<u>1R07</u> <u>Heat Sink Performance</u>

CPS 065-017; Heat Exchanger Performance Trending – Main Steam Isolation Valve Leakage Outboard Cooler (1VY10S)

CPS 065-017; Summary Report Clinton Generic Letter 89-13 Program Report

CPS 065-017; Heat Exchanger Performance Trending – Division 4 Inverter Room Cooler (1VX14S)

CPS 1003.10 ; Program for NRC Generic Letter 89-13

CPS 2602.01; Heat Exchanger Performance of Shutdown Service Water Coolers

CPS 2602.01D008; Shutdown Service Water Pump Room 1A Cooler

CPS 2602.01D008; Shutdown Service Water Pump Room 1A Cooler

CPS 2602.01D009; Shutdown Service Water Pump Room 1B Cooler

CPS 2602.01D009; Shutdown Service Water Pump Room 1B Cooler

CPS 2602.01D017; Division 4 Inverter Room Cooler

CPS 2602.01D017; Division 4 Inverter Room Cooler

CPS 2602.01D028; Main Steam Isolation Valve Leakage Outboard Room Cooler

CPS 2602.01D028; Main Steam Isolation Valve Leakage Outboard Room Cooler

CPS 6952.01D001; Liquid Process Radiation Monitor Contingency Sampling and Analysis Data Sheet

CPS 6952.01D001; Liquid Process Radiation Monitor Contingency Sampling and Analysis Data Sheet

CPS 6952.01D001; Liquid Process Radiation Monitor Contingency Sampling and Analysis Data Sheet

CPS 6952.01D001; Liquid Process Radiation Monitor Contingency Sampling and Analysis Data Sheet

CPS 8199.01; Concrete Expansion Anchor Work

CR 00123025; Shutdown Service Water Piping Degradation

CR 00123504; Shutdown Service Water Silting and Piping Degradation

CR 00123592; Shutdown Service Water Piping below Specified Minimum Wall Thickness

CR 00124374; Residual Heat Removal B Heat Exchanger Tube Degradation

CR 00126583; 1RIX-PR039 B Shutdown Service Water Effluent Monitor Numerous Alert Alarms

CR 00132333; Shutdown Service Water Piping below Specified Minimum Wall Thickness

CR 00140227; 1RIX-PR039 B Shutdown Service Water Effluent Radiation Monitor Spike

CR 00142761; Silt Levels in the Shutdown Service Water and Fire Protection Pump Bays

CR 00145166; Qualitative Analysis of Sponge Balls Indicate Co-60 Activity

CR 00145836; Received Alert Alarm on 1RIX-PR039 B Shutdown Service Water Effluent

CR 00148468; Heat Exchanger Performance Test Evaluation

CR 00149662; Residual Heat Removal Heat Exchanger Tube Plugging

CR 00152179; Residual Heat Removal A Heat Exchanger Assessment in Engineering Change Evaluation

CR 00154028; Piping Flushes of Dead Legs of Shutdown Service Water Piping Created by Condition Report 74732

CR 00156245; Eddy Current Sizing Error on Residual Heat Removal Heat Exchanger

CR 00157115; Lower than Allowable Flows Through Main Steam Isolation Valve Room Cooler

CR 00157361; Water Bubbling Out of Shutdown Service Water Division 1 Carrier Water Guard Pipe

CR 00159157; Unexpected Results from Residual Heat Removal B Electro- Chemical Probe

CR 00160302; Emergency Diesel Generator Flushing Shutdown Service Water Piping Developed Small Leak

CR 00167814; Dead Leg Inspection (Enhancement Condition Report)

CR 00173046; 1RIX-PR039 B Shutdown Service Water Effluent Went into Alert

CR 00176374; Residual Heat Removal Heat Exchanger Electro-Chemical Probe Value

CR 00176531; Enhancement – Perform Monthly Flushes of Residual Heat Removal Heat Exchangers

CR 00176960; Several Relief Valves on Division 2 Shutdown Service Water Lifted on September 22, 2003

CR 00178272; Heat Exchanger Corrosion Pitting on End Bell at Divider Plate

CR 00182871; Abnormal Water Hammer Noise

CR 00186659; Shutdown Service Water System Piping Wall Thinning

CR 00189152; 1SX12AA Wall Loss Due to Pitting

CR 00199270; Observations in Screen House Tunnel

CR 00199851; Broken Service Water Hanger in Circulating Water Screen House Shutdown Service Water Pipe Tunnel

CR 00199858; Shutdown Service Water Hanger in Circulating Water Screen House Shutdown Service Water Tunnel Has Bent Rod

CR 00199862; 2 Damaged Light Fixtures in Circulating Water Screen House Shutdown Service Water Tunnel

CR 00199884; Damaged Insulation on Shutdown Service Water and Service Water Piping in Circulating Water Screen House Shutdown Service Water Tunnel

CR 00199893; Repair Residual Heat Removal A Heat Exchanger Divider Plate

CR 00208522; Shutdown Service Water Wall Thickness At or Below Minimum Wall

CR 00209054; Handrail Interfering with Insulation on Cross-connect Line Between Divisions 1 and 2 Shutdown Service Water

CR 00209077; Procedural Error Found During Ultimate Heat Sink Inspection*

CR 00209086; 89-13 Procedures Contain Inadequate Instructions for Standard Deviation Acceptance Criteria*

CR 00209240; Gap Observed Behind Baseplate of Tubing Support*

CR 00211005; Excessive Gap Found Behind Shutdown Service Water Hanger Base Plate*

CR 00211161; Small Bore Hangers in Division 2 Shutdown Service Water Pump Room*

EC 343286; Extend Periodicity of Testing for Safety Related Coolers or Heat Exchangers Cooled by Standby Service Water for Generic Letter 89-13

M-1SX01034R; Component Support Assembly

M-1SX01045R; Component Support Assembly

MB-200-096; Nuclear Containment Cooling Coils

MC-135-587; Nuclear Containment Cooling Coils

MC-136-413; Nuclear Containment Cooling Coils

MC-136-414; Nuclear Containment Cooling Coils

MO5-1052; Piping and Instrumentation Drawing, Shutdown Service Water

MO5-1052; Piping and Instrumentation Drawing, Shutdown Service Water

MO5-1052; Piping and Instrumentation Drawing, Shutdown Service Water

MO5-1052; Piping and Instrumentation Drawing, Shutdown Service Water

MO5-1052; Piping and Instrumentation Drawing, Shutdown Service Water

P-1SB01021G; Component Base Plate Support Assembly

R107D-1110303; House Assembly, Cooling Coil Shutdown Service Water Pump Room

R107D-1251024; Cooling Coil House Arrangement, Inverter Room Cooler

VY-45; Updated Performance Evaluation and Criteria for 1VY10S Cooling Coils

VY-45; Performance Evaluation of Ventilation System Cooling Coils under Shutdown Service Water Flow Acceptance Limits

1R08 Inservice Inspection Activities

Procedure No. PDI UT 300; Procedure for Manual Examination of Reactor Vessel Assembly Welds in Accordance with PDI; Version 8; dated December 12, 2003

Procedure No. GE-UT-209; Procedure for Automated UT Examination of Dissimilar Metal Welds and Nozzle to Safe End Welds

Procedure No. GE-PDI-UT-1; PDI Generic Procedure for the UT Examination of Ferritic Pipe Welds; Revision 3

Procedure No. GE-PDI-UT-2; PDI Generic Procedure for the UT Examination of Austenitic Pipe Welds; Revision 3

Procedure No. MT-EXLN-100V3; Procedure for Magnetic Particle Examination (Dry Particle, Color Contrast or Wet Particle, Fluorescent); Revision 0

Procedure No. GE-VT-101; Procedure for VT-1 Examination; Version 1

Procedure No. GE-VT-103; Procedure for VT-3 Examination; Version 2

Procedure No. PT-EXLN-100V3; Procedure for Liquid Penetrant Examination Using Fluorescent and Visible Dye Liquid Penetrant Inspection Methods; Revision 0

Procedure No. VT-CLN-206V5; Procedure for Invessel Visual Inspection (IVVI) of WR 6 RPV Internals; Revision 0

Examination Summary Sheet; Component: Dollar Plate Weld (CH-C-1); Report No. 02-001

Examination Summary Sheet; Component: Recirculation Inlet Nozzle at 240 Degrees (N2G); Report 02-008

Reducer to Pipe Exam (1-FW-1-1-2); Exam No. 113800

Safe-End to Nozzle Exam (N4B-W-1); Exam No. 004940

WPS for BI-Process Welding of Carbon Steel, Revision 12b; CPS No. 8209.50

Generic Class I, II and III Operational Pressure Test; CPS No. 9843.02D001; WO No. 00003604

Radiographic Summary Report; WO No. 3604-15

Section XI Repair and Replacement Data Sheet; Work Document No. 00003604

NIS-2 Form; Report No. RF-8-024

Welding Technique Sheet No. 1 for P-No. Materials without Postweld Heat Treatment and without Impact Testing, Revision 7; CPS No. 8209.50D001

Meridional Weld (CH-MD) @195 Degrees GE Examination Summary Sheet; Report No. IR 09-02

Meridional Weld (CH-MD) @ 135 Degrees GE Examination Summary Sheet; Report No. IR 09-01

Division I, II and III Pipe Wall Thinning Analysis (CC-AA-309-1001 Rev 0); Analysis No. 1 P-M-0513

Section XI Repair and Replacement Data Sheet; Work Document No. 00002653

Head Seam Details Drawing; VPF No. 3653-6; Drawing No. 6

Ultrasonic Examination Indication Report for RHR system weld (1-RH-14-24); Report No. IR 09-23

Magnetic Particle Examination Report for RHR Pipe to Penetration Weld (1-RH-14-30); Report No. IR09-13

UT Examination Volume Coverage in CIRO8 (2002) for R-ISI; AR 00200725

UT Records in C1R08, Risk ISI Category Not Identified; AR 00200733

Reconciliation of NDE Acceptance Criteria Not Performed; AR 00201117

1R11 Licensed Operator Requalification

ESG-LOR-70; “Loss of 1A - Rod Drift - ATWS,”

1R12 Maintenance Effectiveness

Safety Valve Test Data

AR 00155275 Report, Quad Cities, April 16, 2003. 2-0203-3B PORV inadvertently opened at power

AR# 00168622, Cantera, September 5, 2003. Review QC (CR 154275) & LIM (CR 60832) Root cause reports

Exelon Corporation Safety Relief Valves Self Actuation Fleet Extent of Condition/Monitoring Recommendations

Dikkers, Nuclear safety/relief valves, Instruction Manual G471-6/125.04:10
CPS 8215.02C001; SRV Removal and Installation Checklist

CPS 8216.02; Safety/Relief Valve Removal and Installation

CPS 8216.01; Safety/Relief Valve Maintenance

CPS 8216.01C001; SRV Maintenance Checklist

CPS 8216.03; Safety/Relief Valve Testing/Inspection

Licensee Event Report 265/03-002, "Self-Actuation of Main Steam Relief Valve due to Excessive Leakage Through Pilot Valve Seat." Quad Cities Nuclear Power Station, Unit 2

Quad Cities Nuclear Power Station, Unit 2, NRC Special Inspection Report 50-265/03-06

1R13 Maintenance Risk Assessment

CPS 3503.01; Class 1E swing battery charger 1DC11E feed to safety related bus inspection and testing

WO# 656030; Self test automatic depressurization system - 4. Division 2

WO# 656030; Replacement of automatic depressurization system #4 card
9080.21; Division 1 integrated test

WO# 00442057; DG 1A Integrated Test

AR 00201828 Report; 1E22F005 did not pass HPCS valve operability test 9051.02

1R15 Operability Evaluations

OE 197883; "Division 3 degraded voltage time delay,"

CR 200534; HPCS water hammer during Division 3 integrated testing

ECR# 363628; HPCS System Walkdown

AR 00205913 Report; Question Related to Sound/Noise in the HPCS room

AR 00204692 Report; NRC question from operability evaluation

AR 00201268 Report; 9080.23 - Fill-Vent piping downstream of Inj VLV 1E22-F004

AR 00201144 Report; HPCS System experienced unusually high pressure spike

1R16 Operator Work-Arounds

ATI 123822-12, Loss of NPSH causes air compressor trips (applies to all pumps)

CR 203370, Software problem caused makeup to FWST to stop without notification

CR 671701, 1G36-F029A/30A Strainer blowdown valves leaking by, filling the RWCU BWRT

WO 423037, Valve leaks past seat causing an impairment to SJAE ‘A’ startup

1R17 Permanent Plant Modifications

EC330994 Rev. 1; change boron 10 enrichment in standby liquid control tank WO#489967

1R19 Post Maintenance Testing

WO 452030.01, CPS 9065.02 Secondary Containment Integrity using VG Train ‘A’

CPS 9054.01, RCIC LP and HP testing as PMT for pump rebuild, Revision 42a

WE 529365; MSIV Isolation System Response Time Test

CPS 9015.02; Standby liquid Control Inject operability follow maintenance

WO 531246-10; Replacement/Removal of snubber 1RH090555 RHR common shutdown cooling suction line

CPS 9054.01, RCIC Operability Check, Revision 42a; Reactor Core Isolation Cooling and HP testing as post maintenance testing for pump rebuild

WO 656431, EM Support Vendor/Adjustment of Div-1 16-cylinder Actuator

CPS 9054.01, RCIC Operability Check, Revision 42a

Southwest Fibs Dog. HP-5

S & L Dog. M06-1074, Sheets 1 through 6

N.E.S.-MS-01.3, “Water hammer Transient Evaluation”

1R20 Refueling and Outage Activities

Issue Report 200811; Place the double blade guide in the core

9093.01C002 Multiple CRD WR131638

Issue Report 201087; Removal verification

WO# 665504-2; 74-relay replace for the Division 3 Shutdown Service Water System (1AP31E/20)

CPS 9000.01D001; Reactor Core Isolation Cooling suppression pool level transmitter replacement PMID (0015897202), source range monitor functional check

CPS 9027.01C001; RSP Operability, RHR A checklist

CPS 9000.06; Reactor Coolant and Vessel Metal/Pressure/Temperature Limit Logs

CR# 200811; Incorrect Double Blade Guide Grappled, Event Anatomy MA-AA-721-1001

INR-C1R09-04-01; Dryer Drain Channel 8 Indication, Indication Notification Report

RH-A05; Clearance 03-02

CPS 9861.04; MSIV Local Leak Rate Test (MC-5,6,7,8,)

AR 00202855 Report; Valve Failed PMT 1E22F036

AR 00201749 Report; Higher than expected as-found opening force on MOV 1E22F004

AR 00202638 Report; Intermediate MCR Indication On 1E22-F036

CPS 3007.01; Preparation and Recovery From Refueling Operations

CPS 3007.01.01C005; Operations With A Potential For Draining The Reactor Vessel Checklist

CPS 9813.01D002; CRD Testing

0000339004; SLC Boron Concentration Increase to Support EPU and 24 month cycle operation

00489867; 1B33N026A ED OPS Brief EC 335695, waiting for task 11 ES Approve test results, Task 30 IM Install conduit for EC, Task 34 ED issue CCRDs

489973; 1E12F009 OPS Brief for EC 339006, waiting for: EC Rev 02 @ Assigned, Task 05 EM install actuator, Task 06 ED CCRD updates in ECR, Task 11 ED notify Simulator group, Task 12 EM Votes Test new operator, Task 13 ES Acceptance of Test results

489996; Perform OPS Brief, Waiting for: Task 01 IM install EC 339011, change min flow valve for TDRFPA, Task 03 ES Accept mod testing, Task 08 ED notify Training

658439; FW flow line A transmitter, Waiting on Task 01 to restore flow xmtr to pre-amag span. Issue #145

0000338994; Revise the low wt source tank level trip to be fault tolerant to prevent the loss of wt and subsequent inadvertent scram

0000330706; Replace relief valve 1E12-F055B (8x12 for steam condensing) with a smaller valve to minimize high dose and manhours to test

0000333697; Remote test function component removal on 1E12F041A approved for C1R09

0000333700; Remote test function component removal on 1E21F006 To C1R09, Need to process Rev 1 before C1R09

0000335773; Provide design change to remove the ATWS power supply output fuses, increase the fuse rating to prevent sporadic blowing of the input fuses or relocate the fuses to the power supply inputs

0000337886; Redesign AUX steam hangers 1AS02003R & 1AS02004R to include water-hammer loads

0000335580; Replacement of the reactor recirculation pump motor 1B33-C001A/B lower guide sleeve bearing with a tilting pad bearing

0000339006; Modify 1E12F009, RHR shutdown cooling inboard isolation valve, to resolve opening issues

0000339058; Eliminate seal staging flow (1B33N007B) high alarm for RR pump 1B

0000341362; Normalize FW RX water level channels A, B, and C to full power average

0000341010; Replace existing reactor core isolation cooling (RI) piping (1R135A and 1R136A) of carbon steel material with 2 1/4 CR-1 (chrome-moly) material

1R22 Surveillance Testing

Issue Report #200811; Place the double blade guide in the core

CPS 9093.01C002 Multiple CRD WR#131638

Issue Report #201087; Removal verification

WO# 665504-2; 74-relay replace for the Division 3 Shutdown Service Water System (1AP31E/20)

CPS 9000.01D001; Reactor Core Isolation Cooling suppression pool level transmitter replacement PMID (0015897202), source range monitor functional check

CPS 9027.01C001; RSP Operability, RHR A checklist

Issue Report 200811; Place the double blade guide in the core

CPS 9093.01C002 Multiple CRD WR131638

Issue Report 201087; Removal verification

WO 665504-2; 74-relay replace for the Division 3 Shutdown Service Water System (1AP31E/20)

CPS 9000.01D001; Reactor Core Isolation Cooling suppression pool level transmitter replacement PMID (0015897202), source range monitor functional check

CPS 9027.01C001; RSP Operability, RHR A checklist

CPS 9065.02; Secondary Containment Revision 29

CPS 9065.02D001; Secondary Containment Integrity Data Sheet Revision 29

CPS 9067.01; Standby Gas Treatment System Train Flow/Heater Operability Revision 30

CPS 9067.01D001; SGTS Train Flow/Heater Operability Data Sheet Revision 27

CPS 9080.01; Diesel Generator 1A Operability - Manual and Quick Start Operability Revision 49c

CPS 9052.01D001; LPCS/RHR A Pump & LPCS/RHR A Water Leg Pump Operability Data Sheet Revision 40a

CPS 3506.01; Diesel Generator and support systems

CPS 9609.01; Shutdown Service Water Operability test Division 2 (WO 654249), Revision 27

CPS 9333.40; Division 3 4.16 kV bus under-voltage relay (degraded voltage) functional final test

CPS 9091.02; Refuel Bridge (F15) Crane/Hoist Operability

CPS 9080.21; Division 1 Emergency Diesel Generator integrated testing, Revision 25d;

CPS 9080.23; Diesel Generation 1C-ECCS integrated performance test, Revision 27c;

CPS 9080.25; Diesel Generator 1B Test Mode Override load reject operability, and idle speed override

CPS 9080.23; Diesel Generator 1C-ECCS integrated testing Revision 27c

Issue Report 200534; Unexpected Alarm on 5009-3A, activated seismic recorder

Issue Report 200632; Potential Water & Issue Report PMP 1C unavailable alarm

Issue Report 194427; Load Imbalance between Tandem Engines for 1DG01KA

1R23 Temporary Plant Modifications

LS-AA-104, Exelon 50.59 Review Process

CC-AA-112, Exelon Temporary Configuration Changes

CPS 1019.05, Appendix E, Installation and Cable Routing Requirements for Field Installed Temporary Cables

CPS 50.59 Screening No. CL-2004-S-009

CC-AA-102 Attachment 1A, Design Change Attribute Review,

CPS 1019.05, Control of Transient Equipment/Materials Revision 9

1EP4 Emergency Action Level and Emergency Plan Changes

Clinton Power Station Annex to the Exelon Standardized Emergency Plan; Revision 5

2OS1 Access Control to Radiologically Significant Areas

CR 200048; C1R09 Lesson RP Tech Cuts Head While Surveying RPV Head Flange

CR 200184; Chewing Tobacco Use in the RCA

CR 200209; Prompt Investigation Report - ED (Electronic Dosimeter) Dose Rate Alarm Received During Fuel Transfer

CR 200918; C1R09 Engineering Programs Contractor Leaves ED in PCs

CR 200930; Inadequate RAM Labeling of Contaminated Laundry Bags

CR 201154; Radworker Violation

CR 201244; Radioactive Contaminated Area Not Properly Established

CR 201392; Individual Alarms PCM-1B, Exits Into Clean Area

CR 201460; LPRM Fragment Recovery

RP-AA-460; Controls for High and Very High Radiation Areas

RWP No. 10002826; C1R09 - RR ‘B’ Pump/Motor Replacement

RWP No. 10002827; C1R09 Drywell - SRV Work

RWP No. 10002838; C1R09 Drywell - SRM/IRM Detector Repair/Replace/PMs

RWP No. 10002839; C1R09 Drywell - LPRM Removal Undervessel

RWP No. 10002841; C1R09 Drywell - RT (Reactor Water Cleanup) System Work

RWP No. 10002844; C1R09 Drywell - Tours/Inspections/Surveillance Tests

RWP No. 10002862; C1R09 Dive Suppression Pool to Retrieve Debris

RWP No. 10002866; C1R09 Refuel Floor - Fuel Movement/Core Alts/Vessel ISI-IVVI on 828' Ct and 755' FB

RWP No. 10003781; 2004 NRC Tours and Inspections

2OS2 As Low As Is Reasonably Achievable Planning And Controls (ALARA)

ALARA Plan No. 10002826-01; Reactor Recirculation Pump Replacement Including Pump Motor; Revision 1

ALARA Plan No. 10002827; C1R09 SRV Replacement

ALARA Plan No. 10002838-01; C1R09 Drywell - SRM/IRM Detector Repair/Replace/PMs; Revision 1

ALARA Plan No. 10002839; Drywell - LPRM Removal, Removal/Installation of the BEDS Cutting Unit, Transport of BEDS Cask To/From the Cask Loading Pool and Removal of the LPRM Canister

ALARA Plan No. 10002842; Drywell Scaffold for C1R09

Clinton C1R09 Refueling Outage Collective Radiation Exposure [Spreadsheet Chart]

CR 201183; C1R09 Source Term; dated February 12, 2004 [NRC-Identified Issue]

RP-AA-400; ALARA Program; Revision 2

RP-AA-401; Operational ALARA Planning and Controls; Revision 2

RP-AA-441; Evaluation and Selection Process for Radiological Respirator Use; Revision 2

Survey Index No. 02-04-02-22; Drywell - 723' El. Drywell Basement (C1R08) - Initial Entry Survey

Survey Index No. 02-04-04-53; Drywell - 723' El. Drywell Basement (C1R08) - Pre-Shielding Survey

Survey Index No. 04-02-03; Drywell - 723’ El. Drywell Basement (C1R09) - Pre-Shielding Survey

Survey Index No. 04-02-03-17; Drywell - 723’ El. Drywell Basement (C1R09) -Initial Survey

TEDE ALARA Evaluation 04-15; RWP 10002826

TEDE ALARA Evaluation 04-34; RWP 10002839

TEDE ALARA Evaluation 04-35; RWP 10002826

TEDE ALARA Evaluation 04-50; RWP 10002838

TEDE ALARA Evaluation 04-51; RWP 10002838

TEDE ALARA Evaluation 04-53; RWP 10002827

TEDE ALARA Evaluation 04-61; RWP 10002826

Work-In-Progress Review; RWP 10002826 - WIP No. 1

Work-In-Progress Review; RWP 10002827 - WIP No. 1

Work-In-Progress Review; RWP 10002827 - WIP No. 2

Work-In-Progress Review; RWP 10002838 - WIP No. 1

Work-In-Progress Review; RWP 10002841 - WIP No. 1

Work-In-Progress Review; RWP 10002841 - WIP No. 2

Work-In-Progress Review; RWP 10002841 - WIP No. 3

4OA1 Performance Indicator Verification

Clinton PI Data Summary Report Q4/2003

4OA2 Identification and Resolution of Problems

RHR A Pump Minimum Flow Piping Erosion; AR 00198323

RHR B Pump Minimum Flow Piping Erosion; AR 00196787

Service Water Silting and Piping Degradation ISX25A-4; AR 00123504

Service Water Piping Below the Specified Minimum Wall Thickness; AR 00123592

Bottom Head Drain Line Wall Thinning; AR 00158218

EDG Flushing Service Water Piping Developed Small Leak; AR 00160302

Service Water Piping Wall Loss Due to Erosion (Ref CR 160302); AR 00164827

Service Water System Piping Wall Thinning; AR 00186659

I SX12AA Wall Loss Due to Pitting; AR 00189152

RHR B Pump Minimum Flow Piping Erosion; AR 00196787

4002 Abnormal RPV Level/Loss of Feedwater at Power; AR 00168629

Reactor Scram; AR 00188839; dated December 2, 2003, and the prompt investigation

Empty New Fuel Box Departs W/O Proper Documentation; AR 00194301

Sediment Pond PH Above NPDES Allowable Value; AR 00195216

Incorrect Fuel Bundle Grappled and Lifted; AR 00199043

ED Dose Rate Alarm Received During Fuel Transfer; AR 00200209; dated February 7, 2004, and the prompt investigation dated February 10, 2004

Finger Injury to Mechanical Maintenance Mechanic; AR 00201760; dated February 14, 2004, and the prompt investigation dated February 17, 2004

Near Miss 1E12F009 Valve Started to Operate; AR 00202233; dated February 17, 2004, and the prompt investigation dated February 19, 2004

4OA5 Other

AR# 00179001; Motor Operated Valve Bolt Thread Engagement

WR 114083; generated to perform the inspection in the RHR ‘A’ system outage window the week of October 6, 2003

1E12-F003B; Thread engagement and torque verified by WO# 488163-01 on September 24, 2003. No further action required.

1E12-F048A; Thread engagement and torque verified by review of WO 2629 completed February 6, 2002. No further action required.

1E12-F048B; Thread engagement and torque verified by WO 518504 on September 24, 2003 this valve generated this Condition Report. No further action required.

1E12-F021; Thread engagement and torque verified by WO 201883 on October 25, 2000. No further action required.

1E12-F053A; Task 06 added to C1R09 Work Order 536908. This action will complete in C1R09.

1E12-F053B; Task 04 added to C1R09 Work Order 439830. This action will complete in C1R09.

<u>TI 2515/154</u>

Special Nuclear Materials (SNM) Inventory of July 31, 2003, and update to the inventory for fuel bundles added during refueling outage C1R09, dated February 25, 2004.

Exelon Nuclear Procedure; NF-AA-30; Special Nuclear Material Control Process Description

Exelon Nuclear Procedure NF-AA-300; Special Nuclear Material Control; Revision 4

Exelon Nuclear Procedure NF-AA-300-1000; Special Nuclear Material Control and Periodic Reporting; Revision 2

Exelon Nuclear Procedure NF-AA-310; Special Nuclear Material and Core Component Movement; Revision 6

Exelon Nuclear Procedure NF-AA-330; Special Nuclear Material Physical Inventories; Revision

Exelon Nuclear Procedure NF-AA-600; Integrated Spent Fuel Management

Exelon Nuclear Procedure NF-AA-610; On-Site Wet Storage of Spent Nuclear Fuel; Revision 2

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Is Reasonably Achievable
ASME	American Society of Mechanical Engineers
BI	Barrier Integrity
C1R09	Clinton Power Station's 9th Refueling Outage
CFR	Code of Federal Regulations
CPS	Clinton Power Station
CR	Condition Report
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
ED	Electronic Dosimetry
EP	Emergency Preparedness
ERO	Emergency Response Organization
HPCS	High Pressure Core Spray
HRA	High Radiation Area
IFTS	Inclined Fuel Transfer System
IMC	Inspection Manual Chapter
IRM	Intermediate Range Monitor
LCO	Limiting Conditions for Operations
LPRM	Low Power Range Monitor
MR	Maintenance Rule
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission
OE	Operability Evaluation
OP	Operational Support Center
ORM	Operations Requirements Manual
PARS	Publicly Available Records
PI	Performance Indicator
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RP	Radiation Protection
RWP	Radiation Work Permit
SDC	Shutdown Cooling
SDP	Significance Determination Process
SGTS	Standby Gas Treatment System
SRM	Source Range Monitor
SRV	Steam Relief Valve
SX	Shutdown Service Water
TEMA	Tubular Exchanger Manufacturers Association
TS	Technical Specifications
URI	Unresolved Item
UT	Ultrasonic