UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

August 5, 2008

Mr. Charles G. Pardee
Chief Nuclear Officer and
Senior Vice President
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 INTEGRATED INSPECTION REPORT 05000237/2008-003; 05000249/2008-003

Dear Mr. Pardee:

On June 30, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed report documents the inspection findings, which were discussed on July 8, 2008, with Mr. T. Hanley and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and 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, one self-revealed and two NRC-identified findings of very low safety significance were identified. Two of the findings involved violations of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or severity of a NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. 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, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Dresden Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made 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), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mark A. Ring, Chief
Branch 1
Division of Reactor Projects

Docket Nos. 50-237; 50-249; 72-037
License Nos. DPR-19; DPR-25

Enclosure: Inspection Report 05000237/2008-003; 05000249/2008-003
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Dresden Nuclear Power Station
Plant Manager - Dresden Nuclear Power Station
Regulatory Assurance Manager – Dresden Nuclear Power Station
Chief Operating Officer and Senior Vice President
Senior Vice President - Midwest Operations
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director - Licensing and Regulatory Affairs
Manager Licensing - Clinton, Dresden, and Quad Cities
Associate General Counsel
Document Control Desk - Licensing
Assistant Attorney General
J. Klinger, State Liaison Officer,
Illinois Emergency Management Agency
Chairman, Illinois Commerce Commission

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Letter C. Pardee from M. Ring dated August 5, 2008

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 INTEGRATED INSPECTION REPORT 05000237/2008-003; 05000249/2008-003

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-237; 50-249
License Nos:	DPR-19; DPR-25
Report No:	05000237/2008-003; 05000249/2008-003
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL
Dates:	April 1 through June 30, 2008
Inspectors:	C. Phillips, Senior Resident Inspector D. Meléndez-Colón, Resident Inspector A. Barker, Project Engineer B. Cushman, Reactor Engineer W. Slawinski, Senior Health Physicist D. McNeil, Senior Operations Engineer R. Walton, Operations Engineer
Approved by:	M. Ring, Chief Projects Branch 1 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000237/2008-003, 05000249/2008-003; 04/01/2008 - 06/30/2008; Dresden Nuclear Power Station, Units 2 & 3, Adverse Weather, Fire Protection, and Identification and Resolution of Problems.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three Green findings were identified by the inspectors. Two of the findings were considered NCVs of NRC regulations. 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 safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance with no associated violation of regulatory requirements for the licensee's failure to control loose materials in the protected area. Specifically, on the morning of May 30, 2008, the inspectors identified loose materials that were tornado hazards in direct line of site to the Unit 2 and 3 main transformers and the Unit 3 reserve auxiliary transformer. High winds were forecast for that afternoon. Once notified, the licensee entered the issue into its corrective action program and removed the materials.

  The inspectors concluded that the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on September 20, 2007, because, if left uncorrected, the finding would become a more significant safety concern. The finding is of very low safety significance because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. (Section 1R01.1)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance that involved a NCV of the Dresden Nuclear Power Station Renewed Facility Operating License, Conditions 2.E and 3.G. Two fire doors failed their periodic functional test to demonstrate that the doors could automatically close and were not declared inoperable and appropriate corrective actions were not taken in a timely manner. The door between auxiliary electric equipment room and the Unit 3 cable tunnel (Door 168) failed its functional test on June 9, 2007, and was not repaired until June 18, 2007. The fire door separating the isolation condenser make-up pumps (Door 2001) failed its functional test on May 9, 2007, and was not repaired until May 23, 2007. As part of the corrective actions, the licensee changed the surveillance test procedure to ensure that the doors would be declared inoperable if the test failed in the future.

  Using IMC 0612, Appendix E, "Examples of Minor Violations," issued on September 20, 2007, the inspectors concluded that this finding was more than minor by reviewing example 5.b, in that the equipment was found in an inoperable condition but was returned to service. The inspectors determined that this issue was of very low safety significance because the doors were in very low traffic areas and the probability of the doors being open if a fire were to occur or that someone would pass through either door during a fire scenario was low. The inspectors determined that this issue affected the cross-cutting area of Human Performance because the licensee failed to provide a complete and accurate surveillance test procedure that reflected actual design and license requirements H.2.(c). (Section 1R05)

Cornerstone: Barrier Integrity

- Green. A self-revealed finding of very low significance was identified involving a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to completely verify the adequacy of design information provided by a vendor. In particular, errors were made by the fuel vendor in the hot shutdown boron concentration values used in emergency operating procedures. The deficiency existed between August 23, 2006, and December 22, 2006. The corrective actions for this finding involved requiring the Exelon Nuclear Fuel division to perform the design analysis reviews for core reloads in the future.

  The inspectors concluded that the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on September 20, 2007, because, if left uncorrected, the finding would become a more significant safety concern. The finding is considered to be of very low safety significance because it was based on a design deficiency that was confirmed by the inspectors not to result in loss of operability. The primary cause of this finding was related to the cross-cutting issue of Human Performance, "Work Practices," because the licensee did not ensure supervisory and management oversight of contractor work activities, such that nuclear safety was supported. (H.4.(c)) (Section 4OA2.1)

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 2

On April 26, 2008, power was reduced to approximately 89 percent electrical output to recover control rod drive (CRD) P06. The unit returned to full power on the same day.

On May 28, 2008, power was reduced to approximately 97 percent electrical output to place CRD L13 out of service for maintenance. The unit returned to full power on the same day.

On June 1, 2008, power was reduced to approximately 72 percent electrical output to perform turbine valve testing, a control rod pattern adjustment, and other activities. The unit returned to full power on the same day.

Unit 3

On April 12, 2008, power was reduced to approximately 77 percent electrical output to perform repairs on turbine control valve #3. The unit returned to full power on the same day.

On May 4, 2008, power was reduced to approximately 72 percent electrical output to perform turbine valve testing, a control rod pattern adjustment, and various other activities. The unit returned to full power on the same day.

On June 28, 2008, power was reduced to approximately 85 percent electrical output to perform an end of cycle control rod pattern adjustment and other various activities. The unit returned to full power on the same day.

1. REACTOR SAFETY

1R01 Adverse Weather (71111.01)

.1 Readiness For Impending Adverse Weather Condition – High Wind Conditions

a. Inspection Scope

Because high winds were forecast in the vicinity of the facility for May 30, 2008, the inspectors reviewed the licensee's overall preparations for the expected weather conditions. The inspectors walked down important outdoor areas within the protected area, in addition to the licensee's emergency alternating current (AC) power systems, because safety-related functions could be affected by, or required as a result of, high winds or tornado-generated missiles. The inspectors focused on the licensee's procedures used to respond to specified adverse weather conditions and toured the plant grounds for loose debris, which could become missiles during a tornado or high winds condition. The inspectors also verified that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into its corrective action program in accordance with station procedures.

This inspection constituted one sample prior to the onset of an adverse weather condition.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) for the licensee's failure to control loose materials in the protected area. Specifically, the inspectors identified materials that were tornado or high wind hazards near the Unit 2 and Unit 3 main and Unit 3 reserve auxiliary transformers. No violation of regulatory requirements occurred.

Description: On May 30, 2008, the inspectors conducted a walkdown of the risk-significant portions of the main and auxiliary power system to assess the licensee's preparations to preclude or minimize potential damage from high winds associated with severe storms or tornadoes. High winds were forecast for that afternoon. During the walkdown, the inspectors identified unsecured materials in direct line of sight to the main and auxiliary transformers. The inspectors concluded that high winds or tornadoes combined with the proximity of the transformers to the unsecured materials increased the potential for damage to the transformers or related electrical equipment.

Analysis: The inspectors determined that the failure of licensee personnel to control material in the protected area near risk-significant equipment was a performance deficiency. Using the guidance contained in IMC 0612, "Power Reactor Inspection Reports." Appendix B, "Issue Disposition Screening," dated September 20, 2007, the inspectors determined that the finding is more than minor because, if left uncorrected, the loose items in the vicinity of the main and auxiliary transformers, would become a more significant safety concern. The inspectors determined that the finding warranted evaluation using the SDP because the finding is associated with an increase in the likelihood of an initiating event.

The inspectors evaluated the finding using IMC 0609, Appendix A, Attachment 0609.04, "Initial Screening and Characterization of Findings," dated January 10, 2008. Using the Phase 1 SDP worksheet table 4a for the Initiating Event Cornerstone, transient initiator contributor, the inspectors determined the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. Therefore, the finding is determined to be of very low safety significance (Green).

Enforcement: The failure to maintain the protected area free of tornado hazards was not an activity affecting quality subject to 10 CFR Part 50, Appendix B, nor was a procedure required by license conditions or Technical Specifications (TSs) violated. Therefore, while a performance deficiency existed, no violation of regulatory requirements occurred. The inspectors informed the licensee of the concern and the licensee took corrective action to clean the areas identified by the inspectors. The licensee also commenced a walkdown of outside areas within the protected area to address extent of condition

This is considered a finding of very low safety significance **(FIN 05000237/2008003-01; 05000249/2008003-01)**. The licensee included this finding in its corrective action program as IR 793952.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Unit 2 standby liquid control;
- Unit 3 Division I low pressure coolant injection/containment cooling water; and
- Unit 3 standby liquid control.

The inspectors selected these systems based on their risk-significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Updated Final Safety Analysis Report (UFSAR), Technical Specification (TS) requirements, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

These activities constituted three partial system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Fire Zone 11.1.1, “Unit 3 Reactor Building, Southwest Corner Room,” elevation 476’;

- Fire Zone 11.1.2, “Unit 3 Reactor Building, Southeast Corner Room,” elevation 476’;
- Fire Zone 11.2.3, “Unit 2 Reactor Building, High Pressure Coolant Injection Pump Room,” elevation 476’; and
- Fire Zone 8.2.2B, “Unit 3 Turbine Building, Containment Cooling Service Water Pumps,” elevation 495’.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee’s fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant’s Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant’s ability to respond to a security event. Using the documents listed in the Attachment, the inspectors verified 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 fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee’s corrective action program. Documents reviewed are listed in the Attachment to this report.

These activities constituted four quarterly fire protection inspection samples as defined in Inspection Procedure 71111.05-05.

b. Findings

Introduction: The inspectors identified a NCV of Conditions 2.E and 3.G.of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance (Green).

Description: Two fire doors failed their periodic functional test to demonstrate that the doors could automatically close and were not declared inoperable and appropriate corrective actions were not taken in a timely manner:

The fire door between the auxiliary electric equipment room and the Unit 3 cable tunnel (Door 168) failed its functional test on June 9, 2007, due to rust and calcium deposits built up on the floor blocking the closing. It was repaired on June 18, 2007. Issue Report (IR) 638687 incorrectly stated that as long as the door could be latched, it remained operable.

The fire door separating the isolation condenser make-up pumps (Door 2001) failed its functional test on May 9, 2007, and was not repaired until May 23, 2007. Issue Report 627250 incorrectly stated that the automatic closure function of single swing doors was not required for operability.

The licensee performed the functional test using procedure DFPS 4175-07, “Fire Door/Oil Spill Barrier Surveillance,” Revision 21. Section H.3.c. of this procedure required, as part of the functional test, that the door would close on its own and stated that if a door failed to fully close then the door shall be considered inoperable. A note in the same section of the procedure incorrectly stated that a door that failed to close may be considered operable if it is maintained in the fully closed position. Without an automatic closing device, locking the door, or maintaining a watch on the door it cannot be assured of being in a fully closed position.

The inspectors discussed this with the licensee and the licensee's position at the time was that as long as the fire doors could be manually closed, no fire watch was required and the door was not considered inoperable. The inspectors disagreed with the licensee's decision that the door could remain operable as long as the door could be manually closed. The licensee later agreed with the inspector's interpretation of the requirements.

The Technical Requirements Manual (TRM), Section TSR 3.7.n, required a functional test of fire doors to be performed every 18 months. The TRM bases, B 3.7.n, stated that fire doors are considered functional when the observed condition is the same as the as-designed condition. The as-designed condition was described by the NRC position on fire doors as stated in the Updated Fire Hazards Analysis, Volume 1, page 5.4-5, Amendment 16, as that door openings should be protected with tested and rated doors and such doors should be closed and locked or armed with alarm capability with annunciation in the control room. The licensee's justification for non-compliance with that requirement referred to National Fire Protection Association (NFPA) 80. The licensee was committed to NFPA 80 – 1975. The requirement for metal swinging fire doors per paragraph 3-10.1 of NFPA 80 - 1975, was that the "doors shall be equipped with self-closing or automatic closing devices to ensure that they will be closed and latched at the time of the fire."

The required actions per TRM 3.7.n for an inoperable fire barrier mandates that if one or more fire rated assemblies or sealing devices is inoperable, then the licensee must establish a continuous fire watch within one hour; OR verify the operability of fire detectors on one side of the door AND establish a dedicated roving fire watch AND restore the inoperable fire rated assembly to operable status in seven days; OR establish a dedicated continuous fire watch AND prepare a corrective action program report. The licensee took none of the above actions because the door was declared operable.

<u>Analysis</u>: The inspectors determined that the failure to declare the fire doors inoperable, verify the fire detectors, establish the appropriate fire watches, or repair the doors within seven days was a performance deficiency warranting a significance evaluation. Using IMC 0612, Appendix E, "Examples of Minor Violations," issued on September 20, 2007, the inspectors determined that this finding was more than minor by reviewing example 5.b, in that the equipment was found in an inoperable condition but was returned to service. The inspectors determined that this issue also affected the cross-cutting area of Human Performance because the licensee failed to provide a complete and accurate surveillance test procedure that reflected actual design and license requirements. H.2.(c)

The inspectors completed a Phase 1 significance determination of this issue using IMC 609, "Significance Determination Process," Appendix A, Attachment 0609.04, dated January 10, 2008. The inspectors determined that the finding affected fire protection defense-in-depth strategies, therefore, the inspectors referred to IMC 0609, Appendix F, “Fire Protection SDP.” The inspectors assigned a degradation rating to the barrier as LOW per Step 1.2. The doors were in very low traffic areas and the probability of the doors being open if a fire were to occur or that someone would pass through either door during a fire scenario was low. Therefore. per Step 1.3.1.1 the finding was screened as GREEN.

<u>Enforcement</u>: The inspectors determined that the licensee’s failure to declare the fire doors inoperable, verify the fire detectors operable, establish the appropriate fire watches, or repair the doors within seven days was a violation of Dresden Nuclear Power Station Renewed Operating License. License conditions 2.E and 3.G of the Unit 2 and Unit 3, respectively, Dresden Nuclear Power Station Renewed Facility Operating Licenses state, in part, that, “The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility....” Section 9.5.1, “Fire Protection System,” of Dresden UFSAR states that, “The design bases, system descriptions, safety evaluations, inspection and testing requirements, NFPA conformance reviews, personnel qualifications, and training are described in Reference 1.”

Section 9.5.10, “References,” of Dresden UFSAR, reference 1, lists, “Dresden Units 2 and 3 Fire Protection Reports,” Volumes 1 through 5, and, “Fire Protection Program Documentation Package,” Volumes 1 through 13, as the documents to follow for compliance with the fire protection program.

The NRC position on fire doors as stated in the Dresden Units 2 and 3 Fire Protection Reports, Volume 1, “Updated Fire Hazards Analysis,” page 5.4-5, Amendment 16, is that door openings should be protected with tested and rated doors and such doors should be closed and locked or armed with alarm capability with annunciation in the control room. The licensee’s justification for non-compliance with that requirement referred to NFPA 80. The licensee was committed to NFPA 80 – 1975. The requirement for metal swinging fire doors per paragraph 3-10.1 of NFPA 80 - 1975, was that, “doors shall be equipped with self-closing or automatic closing devices to ensure that they will be closed and latched at the time of the fire.”

The requirement to verify that the fire doors auto close is found in TRM, Section TSR 3.7.n, by a functional test of fire doors to be performed every 18 months. Surveillance test procedure DFPS 4175-07, “Fire Door/Oil Spill Barrier Surveillance,” Revision 21 implemented the TRM requirement.

Contrary to the above, the door between the auxiliary electric equipment room and the Unit 3 cable tunnel (Door 168) failed its functional test DFPS 4175-07 on June 9, 2007, and was not declared inoperable and appropriate corrective actions were not taken until Door 168 was repaired on June 18, 2007.

The fire door separating the isolation condenser make-up pumps (Door 2001) failed its functional test DFPS 4175-07 on May 9, 2007, and was not declared inoperable and appropriate corrective actions were not taken until Door 2001 was repaired on May 23, 2007.

This issue was entered in the licensee’s corrective action program as IR 786178. Corrective actions by the licensee included revising procedure DFPS 4175-07 to require declaring doors that do not close automatically inoperable. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section VI.A.1 or the NRC Enforcement Policy. **(NCV 05000237/2008003-02; 05000249/2008003-02)**

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the UFSAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. The specific documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant area(s) to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- Unit 2 east and west reactor building corner rooms

This inspection constitutes one internal flooding sample as defined in Inspection Procedure 71111.06-05.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On April 21, 2008, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;

- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

This inspection constitutes one quarterly licensed operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11B)

.1 Facility Operating History (71111.11B)

Completion of Sections .2 through .9 constitutes one biennial licensed operator requalification inspection sample as defined in Inspection Procedure 71111.11B.

a. Inspection Scope

The inspectors reviewed the plant's operating history from May 2006 through April 2008 to identify operating experience that was expected to be addressed by the Licensed Operator Requalification Training (LORT) program. The inspector verified that the identified operating experience had been addressed by the facility licensee in accordance with the station's approved Systems Approach to Training (SAT) program to satisfy the requirements of 10 CFR 55.59(c). The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Licensee Requalification Examinations

a. Inspection Scope

The inspectors performed an inspection of the licensee's LORT test/examination program for compliance with the station's SAT program which would satisfy the requirements of 10 CFR 55.59(c)(4). The reviewed operating examination material consisted of three operating tests, each containing two dynamic simulator scenarios and five or six Job Performance Measures (JPMs). The written examinations reviewed consisted of three written examinations, each including a Part A, Plant and Control Systems and Part B, Administrative Controls/Procedure Limits. Each examination contained approximately 35 questions. The inspectors reviewed the annual requalification operating test and biennial written examination material to evaluate general quality, construction, and difficulty level. The inspectors assessed the level of examination material duplication from week-to-week during the current year operating test. The examiners assessed the amount of written examination material duplication from week to-week for the written examination administered in 2007. The inspectors reviewed the methodology for developing the examinations, including the LORT program 2-year sample plan, probabilistic risk assessment insights, previously identified operator performance deficiencies, and plant modifications. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.3 Licensee Administration of Requalification Examinations

a. Inspection Scope

The inspectors observed the administration of a requalification operating test to assess the licensee's effectiveness in conducting the test to ensure compliance with 10 CFR 55.59(c)(4). The inspectors evaluated the performance of one crew in parallel with the facility evaluators during four dynamic simulator scenarios and evaluated various licensed crew members concurrently with facility evaluators during the administration of several JPMs. The inspectors assessed the facility evaluators' ability to determine adequate crew and individual performance using objective, measurable standards. The inspectors observed the training staff personnel administer the operating test, including conducting pre-examination briefings, evaluations of operator performance, and individual and crew evaluations upon completion of the operating test. The inspectors evaluated the ability of the simulator to support the examinations. A specific evaluation of simulator performance was conducted and documented in the section below titled, "Conformance with Simulator Requirements Specified in 10 CFR 55.46." The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.4 Examination Security

a. Inspection Scope

The inspectors observed and reviewed the licensee's overall licensed operator requalification examination security program related to examination physical security (e.g., access restrictions and simulator considerations) and integrity (e.g., predictability and bias) to verify compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The inspectors also reviewed the facility licensee's examination security procedure, any corrective actions related to past or present examination security problems at the facility, and the implementation of security and integrity measures (e.g., security agreements, sampling criteria, bank use, and test item repetition) throughout the examination process. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.5 Licensee Training Feedback System

a. Inspection Scope

The inspectors assessed the methods and effectiveness of the licensee's processes for revising and maintaining its LORT Program up to date, including the use of feedback from plant events and industry experience information. The inspectors reviewed the licensee's quality assurance oversight activities, including licensee training department self-assessment reports. The inspectors evaluated the licensee's ability to assess the effectiveness of its LORT program and the licensee's ability to implement appropriate corrective actions. This evaluation was performed to verify compliance with 10 CFR 55.59(c) and the licensee's SAT program. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.6 Licensee Remedial Training Program

a. Inspection Scope

The inspectors assessed the adequacy and effectiveness of the remedial training conducted since the previous biennial requalification examinations and the training from the current examination cycle to ensure that the licensee addressed weaknesses in licensed operator or crew performance identified during training and plant operations. The inspectors reviewed remedial training procedures and individual remedial training plans. This evaluation was performed in accordance with 10 CFR 55.59(c) and with respect to the licensee's SAT program. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.7 Conformance With Operator License Conditions

a. Inspection Scope

The inspectors reviewed the facility and individual operator licensees' conformance with the requirements of 10 CFR Part 55. The inspectors reviewed the facility licensee's program for maintaining active operator licenses and to assess compliance with 10 CFR 55.53(e) and (f). The inspectors reviewed the procedural guidance and the process for tracking on-shift hours for licensed operators and which control room positions were granted watch-standing credit for maintaining active operator licenses. The inspectors reviewed the facility licensee's LORT program to assess compliance with the requalification program requirements as described by 10 CFR 55.59(c). Additionally, medical records for twelve licensed operators were reviewed for compliance with 10 CFR 55.53(I). The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.8 Conformance With Simulator Requirements Specified in 10 CFR 55.46

a. Inspection Scope

The inspectors assessed the adequacy of the licensee's simulation facility (simulator) for use in operator licensing examinations and for satisfying experience requirements as prescribed in 10 CFR 55.46, "Simulation Facilities." The inspectors also reviewed a sample of simulator performance test records (i.e., transient tests, malfunction tests, steady state tests, and core performance tests), simulator discrepancies, and the process for ensuring continued assurance of simulator fidelity in accordance with 10 CFR 55.46. The inspectors reviewed and evaluated the discrepancy process to ensure that simulator fidelity was maintained. Open simulator discrepancies were reviewed for importance relative to the impact on 10 CFR 55.45 and 55.59 operator actions as well as on nuclear and thermal hydraulic operating characteristics. The inspectors conducted interviews with members of the licensee's simulator staff about the configuration control process and completed the IP 71111.11, Appendix C, checklist to evaluate whether or not the licensee's plant-referenced simulator was operating adequately as required by 10 CFR 55.46(c) and (d). The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.9 Annual Operating Test Results

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the individual JPM operating tests, and the simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee from April 2008 through June 2008 as part of the licensee's operator licensing requalification cycle. These results were compared to the thresholds established in IMC 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," and Inspection Procedure 71111.11, "Licensed Operator Requalification Program." The documents reviewed during this inspection are listed in the Attachment. Completion of this section constitutes one biennial licensed operator requalification inspection sample as defined in Inspection Procedure 71111.11B.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- Unit 2 instrument air and;
- Unit 3 instrument air

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

This inspection constitutes two quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Unit 2 standby liquid control pump out-of-service;
- Unit 3 Division I, low pressure coolant injection/containment cooling water out-of-service;
- Unit 3 isolation condenser out-of-service; and
- Unit 3 Division II, low pressure coolant injection/containment cooling water out-of-service.

These activities were selected based on their potential risk-significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

These activities constituted four samples as defined by Inspection Procedure 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Op-Evaluation 08-002, “HPCI [high pressure coolant injection] 2-2301-8, MOV [motor operated valve] Degraded Resistance Values for Armature Winding Circuit EC [engineering change] 369934”;
- Engineering Change 370122, Revision 00, “Degraded Thermal Performance of the 3B LPCI [low pressure coolant injection] Hx [heat exchanger]; Op-Evaluation 08-003, Revision 00;“

- Issue Report 696567, “Incorrect Methodology In Overthrust Calc [calculations] for MOV [motor operated valve] 3-13-4;
- Engineering Change 370368, Revision **00**, “Limitations on CCSW [containment cooling service water] Flow Rate Due to Increase In Containment Overpressure - EPU [extended power uprate] Issue, Op-Evaluation 08-004;
- Engineering Change 370368, Revision **01**, “Limitations on CCSW Flow Rate Due to Increase In Containment Overpressure – EPU Issue, Op-Evaluation 08-004;
- Engineering Change 370368, Revision **02**, “Limitations on CCSW Flow Rate Due to Increase In Containment Overpressure – EPU Issue, Op-Evaluation 08-004; and
- Engineering Change 370368, Revision **03**, “Limitations on CCSW Flow Rate Due to Increase In Containment Overpressure – EPU Issue, Op-Evaluation 08-004.

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee’s evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment.

This inspection constitutes seven samples as defined in Inspection Procedure 71111.15-05

b. Findings

Introduction: The inspectors identified an unresolved item regarding the corrective actions implemented to prevent the blocking of the 3B LPCI heat exchanger tubes by relic clamshells laying on the inlet tubesheet of the 3B LPCI heat exchanger which caused the degraded thermal performance of the heat exchanger.

Description: During the most recent Unit 3 refueling outage, D3R19 (November 2006), a large amount of relic clamshells were discovered laying on the inlet tubesheet of the 3B LPCI heat exchanger. The cause appeared to be clam shells breaking loose from the walls in Bay 13 due to running containment cooling service water pumps while performing the quarterly HPCI system surveillance test on September 16, 2006. At that time it was unknown whether the condition of the 3B LPCI heat exchanger was degraded to a point that it could not perform its intended function. Approximately 25 percent of the inlet tubes (300 tubes) had some degree of tube blockage.

Design conditions for the LPCI heat exchanger assume a 95 degree F inlet water temperature and a heat removal capability of 71 MBtu/hr. Further review determined that the inlet water temperature required to remove 71 MBtu/hr, under this degraded condition, was about 85.5 degrees F. The licensee determined that the inlet water temperature had been less than 85.5 degrees F for the time period between September 2006 and D3R19; therefore 3B LPCI heat exchanger was determined to have been operable during that time. One of the corrective actions implemented to prevent the blocking of the 3B LPCI heat exchanger tubes by relic clamshells was to return to quarterly cleanings of crib house Bay 13. This event was documented in IR 556633.

On March 20-21, 2008, thermal performance data was collected for the 3B LPCI heat exchanger. Data reduction was performed and the initial analysis indicated that the 3B LPCI heat exchanger thermal performance, when extrapolated to design conditions (95 degrees F inlet water temperature), was approximately 0.6 percent below the UFSAR value (70.586 vice 71 MBtu/hr). The cause of the degradation was unknown, but the most probable fouling mechanism was either macro (debris that block tubes) or micro (microbial/slime that inhibit heat transfer within the tubes). Further evaluation determined that with a heat removal capability of 70.586 MBtu/hr the maximum allowable inlet water temperature for three months and six months from the original test was 93.5 degrees F and 92.5 degrees F, respectively.

On April 21, 2008, the inlet tubesheet of the 3B LPCI heat exchanger was examined using a boroscope. The boroscope inspection found that approximately 10 percent of the inlet tube openings were covered with clams. As a result, the heat exchanger was opened on May 12, 2008. The as-found inspection confirmed that approximately 10 percent of the inlet tubes openings were covered with clams and, in addition, that approximately 80 percent of the inlet tubes had some degree of tube blockage. One tube could not be cleared of debris and had to be plugged.

The level of macrofouling increases when the system is operated. The source of macrofoulants (relic clamshells) was Bay 13. Per Parameter Requirement (PMRQ) 243-04, “D2/3 Qtr PM Insp/Cln Fire Pump Bay/Dwnstrm Screen with Diver,” Bay 13 is cleaned every quarter to ensure the introduction of macrofoulants into the system is minimized. In April 2007, the licensee decided not to perform the cleaning of Bay 13 due to personnel safety issues that affected the divers performing the cleaning despite the increase in clamshells and did not clean the bay until July 2007.

At the end of the inspection period, the licensee was still working on an equipment apparent cause evaluation report to evaluate all causal factors related to this issue. The due date of this evaluation is July 25, 2008 (IR 776598). The inspectors considered the determination of the reason the 3B LPCI heat exchanger became plugged after corrective actions were previously identified to be an unresolved item pending inspector review of licensee evaluation efforts. **(URI 05000249/2008003-03)**

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification:

- Temporary configuration change package 363466, “Install Test Equipment to Monitor Reactor Pressure Vessel (RPV) Level Sensing Line Pressure Variations Per the RPV Level Deviation Troubleshooting Plan,” Revision 1.

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, the UFSAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system. The inspectors also compared the licensee’s information to operating experience information to ensure that lessons learned from other utilities had been incorporated into the licensee’s decision to implement the temporary modification. The inspectors, as applicable, performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations, engineering, and training personnel to ensure that the individuals were aware of how extended operation with the temporary modification in place could impact overall plant performance.

This inspection constitutes one temporary modification sample as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance (PM) activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Unit 3 Isolation Condenser Breaker WO 99145895-01, “D3 6Y[year] PM [preventative maintenance] INSP [inspection] 480V MCC [motor control center] BKR [breaker] MOV [motor operated valve] 3-1301-4, Group 2”;
- Engineering Change # 371356, “3B LPCI Heat Exchanger June 26, 2008 Thermal Performance Test”
- Unit 3 Isolation Condenser Valve, WO 99145895-03, “OP [operations] PMT verify Proper Operation Upon Completion of PM [preventative maintenance]”; and
- Work Order 1135864-01, “Unit 3 EDG [emergency diesel generator] Output Breaker Tripped Open During Surveillance.”

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion), and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes four samples as defined in Inspection Procedure 71111.19-05.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- WO 1114938, "D2 1M [month] TS Reactor Low Pressure (350 psig) ECCS [emergency core cooling system] Permissive Calibration";
- WO 934359-01, "D3 24 Month Technical Specification Isolation Condenser Auto-Actuation Surveillance"; and
- WO 671744-01, "U3 2Year Preventive Maintenance Baker Test LPCI Motor 3-1502-C."

The inspectors observed in-plant activities and reviewed procedures and associated records to determine whether: any preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; as left setpoints were within required ranges; the calibration frequency was in accordance with TS, the UFSAR, procedures, and applicable commitments; measuring and test equipment calibration was current; test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied; test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used; test data and results were accurate, complete, within limits, and valid; test equipment was removed after testing; where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of the safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the Attachment.

This inspection constitutes three routine surveillance testing samples as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

.2 Inservice Testing Surveillance

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- WO 01106603, "D3 Quarterly TS 3B SBLC Pump Test for IST [inservice surveillance test]."

The inspectors observed activities and reviewed procedures and associated records to determine whether: any preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; as left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the UFSAR, procedures, and applicable commitments; measuring and test equipment calibration was current; test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied; test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used; test data and results were accurate, complete, within limits, and valid; test equipment was removed after testing; where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers Code, and reference values were consistent with the system design basis; where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of its safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the Attachment.

This inspection constitutes one inservice inspection sample as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of two routine licensee emergency drills on April 16, 2008, and on April 21, 2008, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the technical support center on April 16 and in the simulator on April 21 to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critiques to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critiques and to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. As part of the inspection, the inspectors reviewed the drill packages and other documents listed in the Attachment to this report.

This inspection constitutes two samples as defined in Inspection Procedure 71114.06-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's occupational exposure control cornerstone performance indicators (PIs) to determine whether the conditions resulting in any PI occurrences had been evaluated, and whether identified problems had been entered into the corrective action program for resolution.

This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed licensee controls and surveys in the following radiologically significant work areas within radiation areas, high radiation areas and airborne radioactivity areas in the plant to determine if radiological controls including surveys, postings and barricades were acceptable:

- Unit 2/3 Turbine Deck;
- Unit 2/3 Reactor Buildings (various areas) including the refueling floor;
- Radwaste Building (various areas); and
- Radwaste Concentrator/Condenser Room.

This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

The inspectors reviewed the radiation work permits (RWPs) and work packages used to access these 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 determine if they were aware of the actions required when their electronic dosimeters noticeably malfunctioned or alarmed.

This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

The inspectors walked down and surveyed (using an NRC survey meter) portions of the areas listed above to verify that the prescribed RWP, procedure, and engineering controls were in place, that licensee surveys and postings were complete and accurate, and that air samplers were properly located.

This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

The inspectors reviewed RWPs for airborne radioactivity areas or for work areas with the potential for generating airborne radioactivity given the work scope to assess the adequacy of engineering controls (e.g., high-efficiency particulate air ventilation system operation, if applicable) and to determine if there was a potential for individual worker internal exposures of greater than 50 millirem committed effective dose equivalent. These work areas were the radwaste concentrator/condenser room and the refuel floor during dry cask work.

Work areas having a history of, or the potential for, airborne transuranics were evaluated to verify that the licensee had considered the potential for transuranic isotopes and provided appropriate worker protection.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

The adequacy of the licensee's internal dose assessment process for internal exposure was assessed. No internal exposures in excess of 50 millirem committed effective dose equivalent occurred since previously reviewed by the NRC in July 2007. However, the inspectors evaluated the internal dose assessment results and associated calculations for those workers that had intakes between July 2007 through April 2008.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

The inspectors also reviewed the licensee's physical and programmatic controls for highly activated and/or contaminated materials (non-fuel) stored within spent fuel or other storage pools to determine whether adequate barriers were in-place to reduce the potential for the inadvertent movement of these materials.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed a sample of the licensee's self-assessments, audits, and Licensee Event Reports, as applicable, related to the access control program to verify that identified problems were entered into the corrective action program for resolution.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

The inspectors reviewed corrective action reports related to access control and high radiation area radiological incidents (issues that did not count as performance indicator occurrences identified by the licensee in high radiation areas less than 1R/hr). 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 NCVs tracked in the corrective action system; and
- Implementation/consideration of risk-significant operational experience feedback.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

The inspectors evaluated the licensee's process for problem identification, characterization, and prioritization and verified that problems were entered into the corrective action program and resolved. For repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution, the inspectors determined whether the licensee's self-assessment activities were also identifying and addressing these deficiencies, as applicable.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

The inspectors reviewed licensee documentation packages for all PI events occurring since the last inspection to determine if any of these PI events involved dose rates greater than 25 R/hr at 30 centimeters or greater than 500 R/hr at 1 meter. Barriers were evaluated for failure and to determine if there were any barriers left to prevent personnel access. Unintended exposures greater than 100 millirem total effective dose equivalent (or greater than 5 rem shallow dose equivalent or greater than 1.5 rem lens dose equivalent) were evaluated to determine if there were any regulatory overexposures or if there was a substantial potential for an overexposure.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

b. <u>Findings</u>

No findings of significance were identified.

.4 <u>Job-In-Progress Reviews and Review of Work Practices in Radiologically Significant Areas</u>

a. <u>Inspection Scope</u>

The inspectors observed the following 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:

- Radwaste Concentrator/Condenser Cleanup Activities and
- Dry Cask (Refuel Floor) Storage Activities.

The inspectors reviewed the radiological job requirements for these activities, including RWP requirements and work procedure requirements, and evaluated the radiological controls, job coverage and radiation worker practices. The inspectors attended the pre-job briefing for the radwaste concentrator work to assess the adequacy of the information exchanged. Job performance was observed with respect to these requirements to assess whether radiological conditions in the work area were adequately communicated to workers through pre-job briefings and postings.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

During job performance observations, the inspectors also evaluated the adequacy of radiological controls, including required radiation and contamination surveys; the radiation protection job coverage, any applicable audio and visual surveillance for remote job coverage; and contamination controls.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

Thee inspectors reviewed the licensee's procedure and practices associated with dosimetry placement and the use of multiple whole body dosimetry and for extremity monitoring in high radiation work areas having significant dose rate gradients to evaluate the application of dosimetry to effectively monitor exposure to personnel and to assess the adequacy of licensee controls. These work areas involved areas where the dose rate gradients were severe thereby increasing the necessity of providing multiple dosimeters or enhanced job controls.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

.5 High Risk-Significant, High Dose Rate-High Radiation Area and Very High Radiation Area Controls

a. Inspection Scope

The inspectors held discussions with the Radiation Protection Manager and supervisors concerning high dose rate-high radiation area and very high radiation area controls and procedures, including procedural changes that had occurred since the last inspection in order to assess whether any procedure modifications substantially reduced the effectiveness and level of worker protection.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

The inspectors discussed with radiation protection supervisors the controls that were in place for special areas that had the potential to become locked high radiation areas (LHRAs) or very high radiation areas (VHRAs) during certain plant operations to determine if these plant operations required communication beforehand with the radiation protection group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

The inspectors conducted plant walkdowns to assess the adequacy of the posting, locking and barrier quality of numerous LHRAs, Level-2 LHRAs (i.e., high dose rate-high radiation areas), and VHRAs.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified

.6 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, of the RWP controls and limits in place, and of the level of radiological hazards present. The inspectors also evaluated that worker performance accounted for these radiological hazards.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

The inspectors reviewed radiological problem reports for which the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause, and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. Problems or issues with planned and taken corrective actions were discussed with the Radiation Protection Manager.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician (RPT) 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 and 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.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

The inspectors reviewed radiological problem reports for which the cause of the event was radiation protection technician error to determine if there was an observable pattern traceable to a similar cause and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

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

.1 Radiological Work Planning and Exposure Performance

a. Inspection Scope

The inspectors reviewed the licensee's dose performance for its November 2007, Unit 2 refueling outage (D2R20), focusing on the following work activities each of which accrued collective dose greater than 4 rem and which exceeded or nearly exceeded the licensee's dose estimates by margins of 50 percent or greater:

- Hydraulic Control Unit System Maintenance;
- Reactor Water Cleanup System Maintenance; and
- Digital Electro-Hydraulic Control Modification.

For each of the activities listed above, the inspectors examined the reasons for inconsistencies between intended (projected) and actual work activity doses as well as time/labor differences, as applicable, to determine if each of these activities were adequately planned and executed.

This inspection constitutes one required sample as defined in Inspection Procedure 71121.02-5.

The inspectors also reviewed the licensee's process and practices for adjusting exposure estimates when unexpected changes in scope, emergent work or other unanticipated problems were encountered which could significantly impact worker exposures. This included determining that adjustments to estimated exposures were based on sound radiation protection and as-low-as-is-reasonably-achievable (ALARA) principles and not adjusted to account for failures to effectively plan or control the work.

This inspection constitutes one required sample as defined in Inspection Procedure 71121.02-5.

The licensee's post-job (work activity) reviews for D2R20 were evaluated to verify that identified problems were entered into the licensee's corrective action program.

This inspection constitutes one optional sample as defined in Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

.2 Monitoring of Declared Pregnant Women and Dose to Embryo/Fetus

a. Inspection Scope

The inspectors reviewed the licensee's monitoring methods and procedures, radiation exposure controls, and the information provided to declared pregnant women to determine if and an adequate program had been established to limit embryo/fetal dose. The inspectors reviewed dose records of declared pregnant women for the current assessment period (through April 2008) to verify that the exposure results and monitoring controls employed by the licensee complied with the requirements of 10 CFR 20.1208 and 20.2106.

This inspection constitutes one required sample as defined in Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151-05)

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures PI for Unit 2 and Unit 3 for the period from the 2^{nd} quarter 2007 through the 2^{nd} quarter 2008, to determine the accuracy of the PI data reported during those periods, Performance indicator definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance, were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports and NRC Integrated Inspection reports for the period of June, 2007 through June, 2008 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Attachment to this report.

This inspection constitutes two safety system functional failures samples as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems Performance Index - Emergency AC Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Emergency AC Power System performance indicator Unit 2 and Unit 3 for the period from 2nd quarter 2007 through the 2nd quarter 2008. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, issue reports, event reports and NRC Integrated Inspection reports for the period of June, 2007 through June, 2008 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Attachment to this report.

This inspection constitutes two MSPI emergency AC power system samples as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 In-depth Review

Identification and Corrective Actions associated with IR 563695, "Non-Conservative Inputs Used in Determination of Emergency Operating Procedure Hot Shutdown Boron Weight."

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors reviewed IR 563695, "Non-Conservative Inputs Used in Determination of Emergency Operating Procedure (EOP) Hot Shutdown Boron Weight," and IR 563267,"DRESDEN 3 CYCLE 20 EOP REPORT ERROR," including the associated Root Cause Report. The inspectors interviewed station nuclear engineering personnel and personnel from Exelon corporation nuclear fuels division.

(2) Issues

The XB-hot-nat value is used to calculate the EOP parameter hot shutdown boron weight (HSBW) that determines the standby liquid control tank hot shutdown injection level during an anticipated transient without scram (ATWS) event.

The Unit 3 XB-hot-nat values reported in Table 1 of Westinghouse Report OPTIMA2-TR058D3-EOP entitled "SVEA-96 OPTIMA2 Fuel Input to the Emergency Operating Procedures" were submitted to Exelon in error in July 2006. The correct values to be included in the report were determined in a Westinghouse calculation, but were not correctly transferred to the report issued to the licensee. This was discovered internally by Westinghouse in November 2006, shortly after Unit 3 was made critical and returned to full power after the D3R19 refueling outage.

The Table 1 values reported were 500 ppm natural boron for a full core of Optima2 fuel and 522 ppm boron for the D3C20 mixed core value. The correct value from the Westinghouse calculation note should have been 718 ppm natural boron for both configurations.

In their review, Dresden station engineering personnel had the opportunity to identify this problem in advance, but failed to do so. On August 23, 2006, the Exelon Design Engineering Manager approved OPTIMA2-TR058D3-EOP revision 0 for use as design analysis. The Owner Acceptance Reviewer (Dresden Design Engineer) assumed, but did not verify, that Exelon Corporate Nuclear Fuels engineering department (NF) had performed a review of all core neutronics calculations as part of either the core design process or the fuel transition process, and checked YES to item 16 of Attachment 2 of CC-AA-309, “Control of Design Analysis.” The significance of this event was that approval of the external design analysis allowed all downstream processes associated with revising the EOP’s to proceed. This was the final barrier within the Exelon configuration control process that could have prevented propagation of the error.

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

The inspectors reviewed IR 563695, “Non-Conservative Inputs Used in Determination of Emergency Operating Procedure (EOP) Hot Shutdown Boron Weight,” and IR 563267,”DRESDEN 3 CYCLE 20 EOP REPORT ERROR,” and the associated root cause report. The inspectors interviewed station nuclear engineering personnel and personnel from Exelon corporation nuclear fuels division.

(2) Issues

Station management did not recognize the issues in the IRs as a problem that needed to be addressed by the station. Therefore, station management neither prioritized nor evaluated this issue.

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The inspectors reviewed IR 563695, "Non-Conservative Inputs Used in Determination of Emergency Operating Procedure (EOP) Hot Shutdown Boron Weight," and IR 563267,"DRESDEN 3 CYCLE 20 EOP REPORT ERROR," and the associated root cause report. The inspectors interviewed station nuclear engineering personnel and personnel from Exelon corporation nuclear fuels division.

(2) Issues

The licensee took no corrective actions in regards to the design analysis review for the above issues at the station level. The design engineer that performed the design analysis stated that there were certain parameters that the engineer checked against a Westinghouse web page that contained background information on the design analysis. Per the design engineer, that action met the intent of the procedure check list on performing a design analysis review. The inspectors questioned whether procedure CC-AA-309, "Control of Design Analysis," was adequate to perform a design analysis review if the guidance in the procedure could be interpreted in more than one way. Station Design Engineering Management personnel stated that Westinghouse had a quality assurance department and that the intent of CC-AA-309 was not to do an in depth analysis of a vendor supplied design analysis but rather to ensure that items specified by the station during the procurement of the design review were actually provided by the vendor. The Quality Assurance Topical Report (NO-AA-10), Revision 80, Section 3, "Design Control," Paragraph 2.9, states, in part, that design work performed by vendors will be reviewed to verify inclusion of inspection, testing and acceptance criteria. This statement appears to agree with the licensee's comments on the extent of vendor design analysis.

This represented one inspection sample as an in-depth review.

d. Findings

Introduction: The inspectors reviewed a self-revealed performance deficiency involving a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to verify the adequacy of design information provided by a vendor. This issue was determined to be of very low safety significance, "Green."

This finding was determined to be self-revealed because:

a) It was not identified by the licensee even after multiple opportunities to do so; and

b) It was identified by a licensee vendor, during the course of work for another station within the same utility, not during a deliberate or focused observation for the work performed for Dresden Station.

Description: In support of the transition to Optima2 fuel for Dresden Unit 3, (Engineering Change (EC) 350134) Dresden Design Engineering received new EOP input parameters in the form of a report titled "SVEA-96 Optima2 Input to the Emergency Operating Procedures" on August 7, 2006. The majority of the values in the report were mechanical properties specific to the SVEA-96 Optima2 fuel design. Also included were shutdown margin calculations for emergency conditions such as ATWS or a failure of control rods to reach the full-in position in the event of a scram.

Despite new results being provided for the shutdown margin calculations, Dresden Design Engineering completed the Owner Acceptance Review (CC-AA-309 Attachment 2) without requesting assistance from Exelon Nuclear Fuels (the design authority for core neutronics calculations) and the report was approved for use with the document number OPTIMA2-TR058D3-EOP. Using the fuel design information contained in this report, Dresden station personnel updated the design analyses and the Dresden-specific EOP to support startup and operation of Dresden Unit 3 Cycle 20 (D3C20) with a mixed core of SVEA-96 Optima2 and GE14 fuel.

On November 27, 2006, Westinghouse informed Dresden Design Engineering that the value for hot shutdown boron concentration (XB-hot-nat) delivered in the EOP input report (OPTIMA2-TR058D3-EOP) was non-conservative for D3C20 operation. Dresden Design Engineering contacted Nuclear Fuels and requested an independent verification of the information received from Westinghouse. Nuclear Fuels reviewed the EOP report and the supporting analysis for the standby liquid control system (SBLC) and concluded that the correct value for XB-hot-nat should be 718 ppm instead of the 500 ppm value that had been provided to Dresden Design Engineering. Nuclear Fuels issued IR 563267 to document the discrepancy between the Westinghouse internal calculation and the report provided to Exelon. Dresden Design Engineering issued IR 563695 to track all required site actions to correct the value for XB-hot-nat.

Timing and lack of documentation complicated follow up on this issue. The error actually occurred in late July 2006. However, Exelon was not notified of the error until November 27, 2006. The root cause investigation was initiated on January 17, 2007, almost six months after the event. As a result, the involved individuals' memories were not completely clear and not all e-mail records from that time were maintained. For example, based on Westinghouse internal emails, on July 20, 2006, sometime between 8:55 and 11:44 a.m., , it was decided to change the hot shutdown boron concentration (XB-hot-nat) in the EOP report from 720 to 500 ppm based on dilution from 287°C to 20°C. According to the content of the Westinghouse internal email, this decision was based on an agreement with Exelon Dresden personnel. However, there is nothing documented as to who at Exelon concurred with that decision, or the specific reason. No one interviewed by the licensee at Exelon or Westinghouse recalled who was involved in the discussion that reached agreement on this approach.

Analysis: The inspectors determined that the failure to verify the adequacy of design information provided by a vendor was a performance deficiency. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on September 20, 2007, because if left uncorrected the finding would become a more significant safety concern. The hot shutdown boron concentration becomes significant during an anticipated transient without scram event. One strategy for this event is to lower reactor water level to lower power level. Once the hot shutdown boron concentration has been added the reactor level can be restored. Without sufficient boron concentration the reactor could become critical again.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix A, Exhibit 1, dated January 10, 2008, and determined that this finding impacted the Mitigating System Cornerstone column. The inspectors answered “Yes” to question #1 under the Mitigating System column on page E1-6. The finding was a design deficiency that was confirmed by the inspectors not to result in loss of operability. Calculations performed by Westinghouse following discovery of the error concluded that 500 ppm of natural boron would result in a hot shutdown condition during D3C20 for the first 1000 MWd/MTU exposure (about 40 calendar days from startup). The corrected value of 718 ppm for XB-hot-nat was incorporated into all Dresden Design Analyses and Dresden EOP’s by December 22, 2006, which was less than 40 days after startup for D3C20. Therefore, the issue screened out as having very low significance (Green). The primary cause of this finding was related to the cross-cutting issue of Human Performance, “Work Practices,” because the licensee did not ensure supervisory and management oversight of contractor work activities, such that nuclear safety was supported. (H.4.(c))

<u>Enforcement:</u> Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions ... The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program … Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses.”

Contrary to the above, on August 23, 2006, Dresden Design Engineering accepted Westinghouse design analysis “SVEA-96 Optima2 Input to the Emergency Operating Procedures,” without checking the adequacy of the design, and the analysis contained design calculation errors. The analysis was approved for use with the document number OPTIMA2-TR058D3-EOP. Using the fuel design reactor physics information contained in this report, Dresden station updated Design Analyses and the Dresden-specific EOP to support startup and operation of Dresden Unit 3 Cycle 20 (D3C20) with a mixed core of SVEA-96 Optima2 and GE14 fuel. The changes to the EOPs included a value for hot shutdown boron concentration delivered in the EOP input report (OPTIMA2-TR058D3-EOP) that was not correct and non-conservative for D3C20 operation. This condition existed until the corrected value of 718 ppm for hot shutdown boron concentration was incorporated into all Dresden Design Analyses and Dresden EOP’s by December 22, 2006, which was less than 40 days after startup for D3C20.

Exelon Nuclear Fuels issued IR 563267 to document the discrepancy between the Westinghouse internal calculation and the report provided to Exelon. Dresden Design Engineering issued IR 563695 to track all required site actions to correct the value for hot shutdown boron concentration.

The corrective actions proposed by Exelon Nuclear Fuels for this issue were extensive but can be summarized by:

(1) Ensuring the involvement of all appropriate organizations when changes to the fuel-related EOP parameters occur.

(2) Ensuring Nuclear Fuels provides input to the fuel vendor regarding the assumptions and acceptance criteria for EOP-related shutdown margin calculations.

(3) Westinghouse to develop task-specific design analysis procedures.

(4) Westinghouse to develop a Human Performance program that is supported by their senior management.

(5) The correct value for hot shutdown boron concentration was inserted into all appropriate procedures.

This violation is being treated as a NCV, consistent with Section VI.A., of the NRC Enforcement Policy. **(NCV 05000237/2008003-04; 05000249/2008003-04)**

.2 Semi-Annual Trending

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” the inspectors performed a review of the licensee’s corrective actions program and associated documents to identify trends that could indicate a more significant safety issue. The inspector’s review was focused on modifications of plant configuration, and consisted of an eight month period from October 2007 through May 2008. The inspector reviewed multiple IRs generated during the time period, in an attempt to identify potential trends. The screening of approximately 200 IRs was accomplished to identify potential barriers that prevented the implementation of plant modifications resulting in a more significant safety issue.

In addition, station performance indicator (PI) PE.04, “Engineering Changes in Implementation,” May 2008 input data and the published PI results were also reviewed. From this review, two engineering changes were selected due to their implementation age being greater than one refueling cycle. The review of the engineering changes was to determine if there was any potential safety impact to the station from the lack of modification implementation.

These activities constituted one inspection sample for semiannual review for trends as defined in Inspection Procedure 71152.

b. Findings

There were no findings of significance identified. The inspector determined that within the areas reviewed, the licensee staff initiated issue reports (IRs) at an appropriate threshold. The IRs reviewed also identified if any repeat or similar condition had occurred in the past. There were no recognized trends or generic barriers that prevented the implementation of plant modifications resulting in a more significant safety issue.

Station performance indicator PE.04, “Engineering Changes in Implementation,” tracks the production, backlog, and aging of requests for engineering changes (EC) that have been delivered by engineering to be implemented in the plant. The time frame of this PI will begin when an EC is approved for implementation, and end when it is implemented and operational in the plant. This PI only measures the station's resource utilization directed toward the reduction of the EC backlog. The PI aging goal of 20 for ECs that are greater than one refueling cycle was established by Exelon corporate. The inspectors' review through the licensee staff determined that the evaluation of the potential safety impact resulting from an EC not being implemented was reviewed through other station processes. As an example, procedure ER-AA-2001, "Plant Health Committee," Revision 11, requires monthly review to determine the need to add or reprioritize configuration changes (CC) on the 20/40 CC list. The 20/40 CC list identifies plant health issues which are configuration changes that include no more than twenty active outage related CCs per unit per refueling outage, and no more than forty active non-outage related CCs.

Engineering change numbers 5771 for the repair of a Unit 2 core spray lower line crack, and 351580 for the installation of a new trip unit on 480 VAC switchgear breaker bus 28 cubicle 5C were selected due to their implementation age being greater than one refueling cycle. The EC approval of 5771 was in September 1997, and the approval of 351580 was in January 2005. The inspectors' review identified that the Unit 2 core spray lower line crack was monitored for growth rate in accordance with BWR VIP guidance during refueling outage reactor internal inspections. The installation of a new trip unit on 480 VAC switchgear breaker bus 28 Cubicle 5C was reviewed through an engineering aging analysis and also was functionally tested through preventive maintenance with the installation of the new trip unit planned for refueling outage D2R21.

.3 Daily Corrective Action Program (CAP) Reviews

a. Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 (Closed) Unresolved item 05000237/2007005-03; 05000249/2007005-03 "Secondary Containment Area Radiation Monitor Range Span"

The inspectors determined that NEI 99-01, Revision 5, eliminated the requirement for an Emergency Action Level for secondary containment infrequently habited area radiation levels completely. The licensee will review Revision 5, when it is approved in 2009 and evaluate the changes for implementation. In addition, the inspector reviewed the basis for the Radioactivity Release Control step in the Emergency Operating Procedures. The basis stated that, “before offsite radioactivity release rate reaches the offsite release rate which requires a General Emergency but only if a primary system is discharging into an area outside the primary and secondary containments, emergency RPV [reactor pressure vessel] depressurization is required.” The inspectors’ concerns were that the need to respond with a technician with a survey meter would inhibit necessary operator actions. After reviewing several scenarios with licensee personnel and the Dresden Emergency Operating Procedure steps the inspectors had no further concerns that requiring a technician to respond with a survey meter would prevent or unnecessarily delay the control room operators from depressurizing the reactor prior to discharging a primary system into an area outside the primary and secondary containments.

This represented one inspection sample. This URI is closed.

.2 <u>(Closed) LER 237/2007-001-00 and LER 237/2007-001-01, “Unit 2 Standby Liquid Control System Tank Inoperable Due to a Small Linear Crack”</u>

On January 18, 2007, at 2110 hours (CST), with Unit 2 at approximately 100 percent power, Dresden Nuclear Power Station control room personnel were notified of a through wall linear crack at the Unit 2 Standby Liquid Control (SBLC) System Tank temperature switch well. The temperature switch well contains a temperature sensor that inputs to indication only and a main control room annunciator. The Unit 2 SBLC System was declared inoperable and TS 3.1.7, “Standby Liquid Control System,” was entered. A review of the possible repair options determined that a repair to the tank could not be completed within the allowable Completion Time of TS 3.1.7. On January 19, 2007, Dresden Nuclear Power Station requested a Notice of Enforcement Discretion (NOED). The NOED requested a temporary 72-hour extension of the Completion Time of Required Action B.1 for TS 3.1.7. The intent of the NOED was to avoid a plant shutdown as a result of compliance with TS 3.1.7, Required Action C.1, which required Unit 2 to be placed in Mode 3 operation (i.e., hot shutdown) on or before 1710 hours on January 19, 2007. The NOED was granted at 0503 hours (CST) on January 19, 2007. The system was restored to operable status by encapsulating the cracked component on January 20, 2007, at 0015 hours (CST) within the time allowed by the NOED.

Although the SBLC Tank was inoperable for a period of time that exceeded TS 3.1.7 allowed Completion Time, the system was restored to operable status within the time allowed by the NOED. The inspectors determined that this issue was not a performance deficiency because the cause of this event was not reasonably within the licensee’s ability to foresee and correct or prevent.

Corrective actions in IR 580658, LER 237/2007-001-00 and LER 237/2007-001-01 were reviewed by the inspectors and no findings of significance were identified.

This represented two inspection samples. These LERs are closed.

.3 <u>Seismic Event on April 18, 2008</u>

a. <u>Inspection Scope</u>

On April 18, 2008, at 4:37 a.m., a seismic event was reported having a magnitude of 5.2 on the Richter scale. According to the United States Geological Survey (USGS)

Earthquake Center, the epicenter of the earthquake was located near West Salem, Illinois. The USGS recorded an aftershock at 10:56 a.m. The seismograph at Dresden Station did not record movement during either event indicating these were not significant. These events were documented as IR 765098, "Seismic Event Documented for Dresden on April 18, 2008."

The inspectors observed and reviewed the licensee's response to both events. The inspectors interviewed multiple personnel and confirmed that the licensee properly classified the event in accordance with emergency action level procedures and verified that the licensee's response was appropriate and in accordance with procedures and training. In addition, the inspectors performed walk downs of both the reactor and turbine buildings. No evidence of damage and/or structural degradation was observed.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Halon System Initiated into Main Computer Room

a. Inspection Scope

On June 3, 2008, at 11:00 a.m., the main control room received various alarms that indicated that the halon suppression system injected into the main computer room. Operations dispatched the fire brigade/incident commander to investigate the main computer room alarms. Although there was an odor of something burnt, there was no obvious evidence of anything burnt or fire. External fans were set up to cool the room while investigating the cause. Once the halon discharged, the halon system for the main computer room was declared inoperable. A fire watch was put in place. The licensee initiated a prompt investigation. The prompt investigation was not able to determine the cause for the halon initiation. This event was documented as IR 782430, "Halon System Initiated into Main Computer Room."

The inspectors observed and reviewed the licensee's response to the event. The inspectors interviewed multiple personnel, verified that the licensee's response was appropriate and in accordance with procedures and confirmed that the event did not meet the threshold for reportability. In addition, the inspectors performed a walk down of the main computer room. No evidence of an actual fire was observed. The inspectors also reviewed the results of the licensee's prompt investigation.

This represented one inspection sample.

b. Findings

No findings of significance were identified

4OA5 Other Activities

.1 Review of Institute of Nuclear Power Operations Reports

The inspectors completed a review of the final report for the Institute of Nuclear Power Operations, October 2007 Evaluation. The onsite inspections were conducted the weeks of September 24 and October 1, 2007. During the review the inspectors did not identify any new safety significant issues.

.2 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 8, 2008, the inspectors presented the inspection results to Mr. T. Hanley, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Radiation protection radiological access control/post outage ALARA program inspection with Mr. D. Wozniak and other licensee staff on May 9, 2008. The inspectors confirmed that none of the potential report input discussed was considered proprietary;
- The results of the licensed operator requalification training program inspection with the Plant Manager, Mr. T. Hanley on May 23, 2008; and
- The licensed operator requalification training annual operating test results with the Licensed Operator Requalification Lead Instructor, Mr. P. O'Connor, via telephone on June 6, 2008.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Wozniak, Site Vice President
T. Hanley, Plant Manager
K. Aleshire, Exelon Corporate Emergency Preparedness Manager
C. Barajas, Operations Director
H. Bush, Radiation Protection Manager
J. Ellis, Regulatory Assurance Manager
D. Galanis, Design Engineering Manager
D. Glick, Shipping Specialist
G. Graff, Operations Training Manager
J. Griffin, Regulatory Assurance - NRC Coordinator
D. Leggett, Nuclear Oversight Manager
M. Overstreet, Lead Radiation Protection Supervisor
C. Podczerwinski, Maintenance Rule Coordinator
P. Quealy, Emergency Preparedness Manager
E. Rowley, Chemistry
R. Rybak, Regulatory Assurance
J. Sipek, Engineering Director
N. Starcevich, Radiation Protection Instrumentation Coordinator
J. Strmec, Chemistry Manager
C. Symonds, Training Director

NRC

M. Ring, Chief, Division of Reactor Projects, Branch 1

IEMA

R. Schulz, Illinois Emergency Management Agency

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened:

05000237/2008003-01 05000249/2008003-01	FIN	Failure to Control Loose Materials in the Protected Area (1R01.1)
05000237/2008003-02 05000249/2008003-02	NCV	Fire Doors Failed Their Periodic Functional Test (1R05)
05000249/2008003-03	URI	Corrective Actions to Prevent the Blocking of the 3B Low Pressure Coolant Injection (LPCI) Heat Exchanger Tubes by Relic Clamshells (1R15)
05000237/2008003-04 05000249/2008003-04	NCV	Failure to Verify the Adequacy of Design Information Provided By a Vendor (4OA2.1)

Closed:

05000237/2008003-01 05000249/2008003-01	FIN	Failure to Control Loose Materials in the Protected Area (1R01.1)
05000237/2008003-02 05000249/2008003-02	NCV	Fire Doors Failed Their Periodic Functional Test (1R05)
05000237/2008003-04 05000249/2008003-04	NCV	Failure to Verify the Adequacy of Design Information Provided By a Vendor (4OA2.1)
05000237/2007005-03; 05000249/2007005-03	URI	Secondary Containment Area Radiation Monitor Range Span (4OA3.1)
237/2007-001-00	LER	Unit 2 Standby Liquid Control System Tank Inoperable Due to a Small Linear Crack (4OA3.2)
237/2007-001-01	LER	Unit 2 Standby Liquid Control System Tank Inoperable Due to a Small Linear Crack (Supplement) (4OA3.2)

Discussed:

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or 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 part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather Protection

- OP-AA-108-107-1001, “Station Response to Grid Capacity Conditions,” Revision 2
- WC-AA-101, “On-Line Work Control Process,” Revision 14
- DOA 6500-12, “Low Switchyard Voltage,” Revision 18
- IR 779949, “NRC Identifies Non-Compliance with WC-AA-101”

1R04 Equipment Alignment

- DOP 1100-M1, “Unit 2 Standby Liquid Control (SBLC) System,” Revision 13
- DOP 1100-E1, “Unit 2 Standby Liquid Control Electrical Checklist,” Revision 04
- DOP 1100-M1/E1, “Unit 3 Standby Liquid Control (SBLC) System,” Revision 14

1R11 Operator Licensing (71111.11B)

- AR# 00740723, “Pre-NRC 71111.11 Inspection FASA Deficiency-Objective 2”
- AR# 00740728, “Pre-NRC 71111.11 Inspection FASA Objective 2 Deficiency”
- AR# 00740730; “Pre-NRC 71111.11 Inspection FASA Objective 2 Deficiency”
- AR# 00740733, “Pre-NRC 71111.11 Inspection FASA Objective 4 Deficiency”
- AR# 00740736, “Pre-NRC 71111.11 Inspection FASA Objective 4 Deficiency”
- AR# 00740739, “Pre-NRC 71111.11 Inspection FASA Objective 5 Deficiency”
- AR# 00740744, “Pre-NRC 71111.11 Inspection FASA Objective 5 Deficiency”
- AR# 00740757, “Pre-NRC 71111.11 Inspection FASA Objective 6 Deficiency”
- AR# 00740767, “Pre-NRC 71111.11 Inspection FASA Objective 7 Deficiency”
- AR# 00740776, “Pre-NRC 71111.11 Inspection FASA Objective 8 Deficiency”
- AR# 00668627, “NOS ID SM/STA CRC Issues”
- AR# 00739989, “TRNG-Simulator DEHC Failed to Reset”
- AR# 0076250, “TRNG-Simulator Tech Specs Not Updated”
- AR# 00729645, “TRNG-Simulator Restoration Delays ILT Examination”
- CR-729050, “Training Dept. Not Set Up for Remediation”
- AR# 00733413, “LORT Weekly Training Failures”
- AR# 00733375, “NOS Identified Errors In Qualification and Drill Records”
- AR# 00754891, “TRNG-Simulator Hardware Issue Delayed Training”
- AR# 00742912, “TRNG-LORT CRC Not Held as Scheduled”
- AR# 00741177, “TRNG Simulator DEHC Issues”
- LER 05000237/2006-04, “MSIV Closure Scram due to Fitting Failure”
- LER 05000237/2007-002, “Unit 2 Scram, Loss of Feedwater, due to Condensate Filter CPU Replacement”
- LER 05000237/2008-01, “Mispositioned Control Rod”
- Simulator Exercise Guide (SEG) OPEX-H 10
- SEG OPEX-AT, Revision 01
- Job Performance Measure (JPM) PO-0300-02, Revision 08

- JPM PO-6600-06, Revision 05
- JPM S-1300-02, Revision 02
- JPM S-6500-06, Revision 03
- JPM S-0500-12, Revision 13
- JPM S-A-03, Revision 01
- SEG OPEX-AG, Revision 03
- SEG OPEX-AN, Revision 01
- SEG OPEX-AM, Revision 01
- SEG OPEX-AQ, Revision 01
- JPM P2-2300-01, Revision 09
- JPM P3-0202-02, Revision 15
- JPM S-1200-03, Revision 01
- JPM S-4400-02, Revision 00
- JPM S-6620-03, Revision 07
- JPM S-A-01, Revision 01
- JPM P2-0300-03, Revision 10
- JPM P2-0500-02, Revision 15
- JPM S-0300-05, Revision 00
- JPM S-1600-06, Revision 00
- JPM S-6600-05, Revision 11
- JPM S-EP-01, Revision 06
- Dresden Station Classroom Sample Plan, dated 8/19/07
- Written Examinations A, B, C & D, E for 2007
- 5-Year Training Plan (2005-2009), Revision 0
- Rolling Examination Report, Revision 0
- 2008 Pre-NRC 71111.11 Inspection Assignment #714175-03, dated 02/25/08
- Training Comprehensive Follow-Up Assignment #499529, dated 10/11/06
- Training Warning Flags/INPO Weaknesses
- Assignment #00559779-04, dated 06/27/07
- Operations Training Record Quality Review
- Assignment #00559790-04, dated 04/27/07
- Assessment of Dresden's Response to INPO SOER 99-01
- Assignment ASSA #629381-04, dated 06/27/07
- SOER 94-1, Non-Conservative Decisions and Equipment Performance Problems Result in a Reactor Scram, Two Safety Injections, and Water Solid Conditions, Recommendation 4 Assignment 629362, dated 06/28/07
- SOER 91-1, Conduct of Infrequently Performed Tests or Evolutions – Recommendation 2, Assignment # 629362, dated 06/22/07
- 2008 Dresden Station Simulator Sample Plan, dated 4/18/08
- Simulator Steady State Testing, dated 11/30/06
- Simulator Transient Test, "TT1, Manual Scram," dated 3/11/07
- Simulator Transient Test, "TT3, MSIV Closure," dated 3/14/07
- Simulator Transient Test, "TT8, LOOP/LOCA," dated 3/14/07
- Simulator Malfunction Test, "A55, Enable Full Core Oscillation," dated 2/27/07
- Simulator Malfunction Test, "FWRV 'A' Lockup Due to Blown Fuse," dated 3/4/04
- Simulator Malfunction Test, "4KV Bus 21 Overcurrent," dated 2/27/07
- Simulator Malfunction Test, "MSIV 203-1A Fast Closure," dated 11/22/06
- Simulator Malfunction Test, "Generator Trip," dated 11/22/06
- Simulator Malfunction Test, "DG 2 Cooling Water Pump Trip," dated 11/22/06
- Normal Plant Evolution Test, "Reactor Scram DGP 02-03," Revision 63
- TQ-AA-301, "Simulator Work Request," dated 5/18/08

- Simulator Configuration Management, Revision 7
- TQ-AA-302, "Simulator Testing and Documentation," Revision 7
- TQ-AA-150, "Operator Training Programs," Revision 0
- OP-AA-105-102, "NRC Active License Maintenance," Revision 9
- OP-OR-101-111-1001, "On-Shift Staffing Requirements," Revision 0

1R12 Maintenance Effectiveness (71111.12)

- UFSAR Section 9.3.1.2, Instrument Air System
- IR 745080, "MR Function Z47-1 Reliability Criteria Exceeded"
- System Health Indicator Program (SHIP) Report – 1st Quarter 2008
- Maintenance Rule Expert Panel Minutes dated 3/26/2008 ((a)(1) Determination)
- Maintenance Rule Expert Panel Minutes dated 5/22/2008 ((a)(1) Action Plan)
- "Failure Analysis of a Lexair Model: 334724-25 Air Operated Valve Component ID: 2-4799-580, Ref.: WO 1011467, Dresden Unit 2"
- MA-AA-716-017, "Equipment Readiness and Reliability," Rev. 1
- ER-AA-310, "Implementation of the Maintenance Rule," Rev. 6
- ER-AA-310-1004, "Maintenance Rule – Performance Monitoring," Rev. 7
- DTS 4700-01, "Sampling Unit 2 (3) Instrument Air," Rev. 8
- IR 788497, "NRC Identifies Several Housekeeping Issues"
- IR 787450, "NRC Identified Unlabeled Freon Tank"

1R13 Maintenance Risk Assessments and Emergent Work Control

- WO 428090-01, "East Piston Leaking Boron Crystals into Drain and Down"

1R15 Operability Evaluations

- IR 755291, "Degraded Thermal Performance of the 3B LPCI Hx"
- IR 776598, "3B LPCI Hx Inspection Results"
- IR 753450, "Intake Temp Recorder Failed"
- IR 763663, "EPU Project did not Evaluate the Effect of Higher Overpressure"
- IR 766830, "NRC Resident Questions Means of Measuring Intake Temperature"
- DOP 1500-02, Rev. 54, "Torus Water Cooling Mode of Low Pressure Coolant Injection System"
- EC# 370381, Rev. 00, "CCSW Flow Rates through the LPCI/CCSW Hxs when Maintaining a 20 psi Higher Pressure than the LPCI System"
- EC# 370382, Rev 00, "CCSW Temperatures to Maintain 71.000 MBtu/hr Heat Exchanger Performance Using CCSW Flow From EC 370381"
- EC# 370466, Rev 00, "CCSW Flow Rates Through the Unit 3 Division I LPCI/CCSW Hx When Maintaining a Higher Pressure Than the LPCI System"
- EC# 370479, Rev 00, "Maximum CCSW Temperature Allowed for the 3B LPCI Hx When Maintaining a 7 psid Between LPCI and CCSW"
- Calc. DRE97-0073, "LPCI/CCSW Heat Exchanger Differential Pressure," Rev. 00
- EC# 370598, Rev. 00, "CCSW Flow Rates Through the LPCI/CCSW HXs when Maintaining a Higher Pressure than the LPCI System"
- EC# 370686, Rev. 00, "LPCI HX Heat Removal Rate Using CCSW Flow Rates from EC 370598"
- EC#370585, Rev. 0, "Evaluate the Impact of LPCI Heat Exchanger Performance at less than 71 MBtu/hr under Design Conditions"

- EC# 370787, Rev. 0, “CCSW Flow Rates Through the Unit 3 Division II LPCI/CCSW Hx when Maintaining a Higher Pressure than the LPCI System”
- EC# 370840, Rev. 0, “LPCI Hx Performance for Past Operability Assessment”
- IR 781234, “CCSW System Historical Operability Review”

1R18 Plant Modifications

- WO 955187-01, “Install Pressure Monitoring Instruments on Reference Leg”
- WO 955187-05, “Post Modification Test Perform VT-2 During Hydro”

1R19 Post Maintenance Testing

- DOS 1300-02, Rev. 15, “Isolation Condenser Valve Operability Check”

1R22 Surveillance Testing

- DIS 1500-01, “Reactor Low Pressure (350 psig) ECCS Permissive,” Revision 24
- DIS 1300-3, “Isolation condenser Initiation and Isolation Logic System Functional Test.” Revision 18
- NES-EIC-17.03, “High Potential Tests,” Revision 0

2OS1 Access Control to Radiologically Significant Areas

- IR 692929, “Visitors in RCA Without Proper Dosimetry”
- IR 720478, “Three Uncontrolled High radiation Area Keys”
- IR 721020, “High Radiation Area Deviation for Box of Spent Fuel Pool Cleanup Equipment”
- IR 770840, “LHRA Doors Need Repairs/Enhancements”
- IR 758409, “RP Posting and Swing gate Moved Without Authorization”
- RP-AA-829, “Operation and Use of Powered Air Purifying Respirators,” Revision 0
- RP-AA-210, “Dosimetry Issue, Usage and Control,” Revision 13
- RP-AA-460, “Controls for High and Very High Radiation Areas,” Revision 13
- RP-AA-460-1001, “Additional High Radiation Exposure Control,” Revision 3
- DFP-0800-39, “Control of Material/Equipment Hanging in Units 2/3 Spent Fuel Pools,” Revision 14
- Self-Assessment Report; Control of Access to Radiologically Significant Areas, dated March 12, 2008
- RWP 10009196, “Associated Surveys and ALARA Plan; Unit 2/3 Radwaste Demineralizer Vault Cleanup,” Revision 0
- RWP 10009120, “Associated Surveys and ALARA Plan; Unit 2/3 Dry Cask Storage Activities,” Revision 0
- TID-2007-003, “Annual Bioassay Program Review,” dated August 20, 2007
- RWP 10007995, “D2R20 Drywell Instrumentation System Maintenance, Revision 1
- RWP 10008006, “D2R20 Drywell Control Rod Drive Support, Revision 1
- RWP 10008015, “Drywell Emergent Work Activities,” Revision 1

2OS2 As-Low-As-Is-Reasonably-Achievable Planning And Controls

- IR 697443, “D2R20 ALARA Tracking”
- RP-AA-270, “Prenatal Radiation Exposure,” Revision 4
- Apparent Cause Report, “D2R20 Exposure Overage,” dated February 21, 2008
- “D2R20 ALARA Outage Report,” undated

- ALARA Post Job Review, “Drywell Emergent Exposure,” dated November 14, 2007
- ALARA Work-In-Progress Reviews, “Drywell Isolation Condenser Valve Rebuild,” dated November 9 & 10, 2007
- ALARA Post Job Review, “Drywell Emergent Insulation Repair,” dated November 12, 2007

<u>4OA3 Event Followup</u>

- IR 782430, “Halon System Initated into Main Computer Room”

<u>4OA3 Event Followup</u>

- IR 782430, “Halon System Initiated into Main Computer Room”

LIST OF ACRONYMS USED

AC	Alternating Current
ALARA	As-Low-As-Is-Reasonably-Achievable
ATWS	Anticipated Transient Without Scram
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CST	Central Standard Time
DFPS	Dresden Fire Protection Surveillance
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
F	Fahrenheit
GE	General Electric
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IR	Inspection Report
IR	Issue Report
JPM	Job Performance Measures
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-site Power
LORT	Licensed Operator Requalification Training
LPCI	Low Pressure Coolant Injection
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
PI	Performance Indicator
PM	Planned or Preventative Maintenance
RP	Radiation Protection
RPT	Radiation Protection Technician
RPV	Reactor Pressure Vessel
RWP	Radiation Work Permit
SAT	Systems Approach to Training
SBLC	Standby Liquid Control
SDP	Significance Determination Process
SSC	Systems, Structures, and Components
TRM	Technical Requirements Manual
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VHRA	Very High Radiation Area
WO	Work Order