January 28, 2003

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION
NRC INSPECTION REPORT 50-237/02-17; 50-249/02-17

Dear Mr. Skolds:

On December 28, 2002, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed report presents the inspection findings which were discussed with Mr. R. Hovey and other members of your staff on January 3, 2003.

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

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green). However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at the Dresden Nuclear Power Station.

Since the terrorist attacks on September 11, 2001, the NRC has issued two Orders (dated February 25, 2002, and January 7, 2003) and several threat advisories to licensees of commercial nuclear power plants to strengthen licensee capabilities, improve security force readiness, and enhance access authorization. The NRC also issued Temporary Instruction 2515/148 on August 28, 2002, that provided guidance to inspectors to audit and inspect licensee implementation of the interim compensatory measures (ICMs) required by the February 25th Order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year (CY) '02, and the remaining inspections are scheduled for completion in CY '03. Additionally, table-top security drills were conducted at several licensees to evaluate the impact of expanded adversary characteristics and the ICMs on licensee protection and mitigative strategies. Information gained and discrepancies identified during the audits and drills were reviewed and dispositioned by the Office of Nuclear Security and Incident Response. For CY '03, the NRC will continue to monitor overall safeguards and security controls and conduct inspections, and will resume force-on-force exercises at selected power plants. Should threat conditions change, the NRC may issue additional Orders, advisories, and temporary instructions to ensure adequate safety is being maintained at all commercial nuclear power plants.

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

Sincerely,

/RA/

Mark Ring, Chief
Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 50-237/02-017;
50-249/02-017

cc w/encl: Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
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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:	50-237/02-017; 50-249/02-017
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	October 1, 2002 through December 28, 2002
Inspectors:	D. Smith, Senior Resident Inspector B. Dickson, Resident Inspector P. Pelke, Reactor Engineer D. Jones, Reactor Engineer H. Gonzalez, Reactor Engineer D. E. Funk Jr., Physical Security Inspector M. Holmberg, Reactor Inspector D. Nelson, Radiation Specialist H. Peterson, Senior Operations Engineer G. O'Dwyer, Reactor Inspector M. Sheikh, Nuclear Safety Intern/Fuel Cycle Inspector R. Zuffa, Illinois Department of Nuclear Safety
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000237-02-017, IR 05000249-02-017; Exelon Generation Company; on 10/1-12/28/2002, Dresden Nuclear Power Station, Units 2 and 3. Heat Sink Performance, Licensed Operator Requalification, and Maintenance Risk Assessments and Emergent Work Control.

This report covers a 3-month period of baseline resident inspection and announced baseline inspections on Temporary Instruction 2515/148, radiation protection, inservice inspection, operator requalification, and heat sink performance. The inspection was conducted by Region III inspectors and the resident inspectors. Two findings involving Non-Cited Violations (NCV) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be 'Green' or be assigned 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 3, dated July 2000.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 55.46(d)(1), "Continued Assurance of Simulator Fidelity" due to the licensee's failure to adequately maintain simulator fidelity for two discrepancies, that had both an actual and potential plant impact. The deficiencies included an incorrect first stage pressure turbine trip reactor scram bypass setpoint and the incorrect operation of the reactor water cleanup (RWCU) room temperature instrument recorder.

This finding was more than minor because the incorrect first stage pressure turbine trip reactor scram bypass setpoint in the simulator had an actual impact on the plant. The incorrect simulator setpoint led to inaccurate training, that subsequently failed to adequately alert the licensed operators of the potential impact of first stage pressure conditions during an actual reactor startup following the Unit 2 power uprate. The lack of simulator fidelity combined with the operators' lack of awareness/attention to the plant effects from the turbine first stage pressure led to an actual reactor scram during the November 7, 2001, reactor startup (see Licensee Event Report 50-237/2001-005-00). Although an actual reactor scram occurred due to high turbine first stage pressure, the finding is of very low safety significance because the discrepancy was on the simulator and the actual plant responded as expected to the high turbine first stage pressure and all safety-related equipment functioned properly. The incorrect operation of the temperature instrument recorder led to an incorrect emergency classification by the Shift Manager during the recent licensed operator requalification annual operating examination. The finding is also of very low safety significance because the discrepancy was on the simulator and the real recorder in the plant functioned properly. Furthermore, no actual plant emergency occurred and there was no actual impact on equipment or personnel safety. (1R11.3)

Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50.65 due to the licensee's failure to perform an adequate assessment of risk during maintenance on the high pressure coolant injection system.

The inspectors concluded that the issue was more than minor since the finding involved a change in risk level from Green to Yellow and, if left uncorrected, could become a more significant safety concern. This conclusion was based on the fact that an adequate assessment of risk could have led to additional management strategies including establishment of protected pathways for redundant mitigating systems.(1R13)

B. Licensee Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 2 began the inspection period at 912 MWe (95 percent thermal power and 100 percent of rated electrical capacity). Operators reduced power to 650 MWe on December 13, 2002, to perform maintenance of the 2C reactor feed pump and the 2A and 2B condensate pumps.

The seventeenth refueling outage on Unit 3 was conducted from October 8-27, 2002. Major work completed during the outage consisted of; modification of steam dryer, replacement of high pressure turbine rotor, electrical modification to the 4KV switchgear, installation of new generator collector rings, modification of drywell steel, upgrades to the main steam and torus attached piping supports, and installation of the second phase of overpower range monitor. The unit became critical on October 26, 2002, and was placed online October 27, 2002. Subsequently, the operators reduced power to 20 percent to facilitate repairs to the 3C feedwater heater on October 30, 2002, and returned the unit to pre-extended power uprate (EPU) full power operations on November 4, 2002. The licensee began EPU testing on November 5, 2002, after reaching 100 percent of the previous reactor power level and completed the testing on November 10, 2002, without any problems. On December 7, 2002, the licensee conducted a forced outage to address the return of increased reactor coolant system leakage. The leak was identified on the same socket weld which was leaking on the 3A reactor recirculation system flow sensing line during the recently completed outage. The unit was placed online on December 11, 2002. Other work performed during the forced outage included replacing a leaking o-ring on control rod drive K-10, repairing the 3B reactor feedwater pump oil deflector, and performing maintenance on the 3B condensate pump seal.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors assessed the licensee's implementation of the station's procedures for the preparation and initiation of cold weather conditions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

The inspectors selected a redundant or backup system to an out-of-service or degraded train, reviewed documents to determine correct system lineup, and verified critical portions of the system configuration. Instrumentation valve configurations and appropriate meter indications were also observed. The inspectors observed various support system parameters to determine the operational status. Control room switch positions for the systems were observed. Other conditions, such as adequacy of housekeeping, the absence of ignition sources, and proper labeling were also evaluated.

The inspectors performed equipment alignment walk-downs of the following systems:

- Unit 2 High Pressure Coolant Injection
- Unit 2 “A” Core Spray
- Unit 3 Isolation Condenser

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

On November 13, 2002, the inspectors observed the fire brigade response to a fire in the turbine building, at the 538' level on motor control center 35-2 and toured plant areas, throughout this inspection period, important to safety to assess the material condition, operating lineup, and operational effectiveness of the fire protection system and features. The review included control of transient combustibles and ignition sources, fire suppression systems, manual fire fighting equipment and capability, passive fire protection features, including fire doors, and compensatory measures. The following areas were walked down:

- Unit 2 Reactor Building, Elevation 476'-6" West Low Pressure Coolant Injection Corner Room (Fire Zone 11.2.1)
- Unit 2 Reactor Building, Elevation 476'-6" High Pressure Coolant Injection Room (Fire Zone 11.2.3)
- Unit 2 Reactor Building, Elevation 545'-6" (Fire Zone 1.1.2.3)
- Fire Doors 12, 52A, 57, 64, and 67 (separate areas)
- Unit 3, Turbine Building, Elevation 534' - 0" DC Panel Room/Feed Water Level Control Station Area (Fire Zone 8.2.6.B)
- Unit 3 Diesel Generator Room, Elevation 517'-6" (Fire Zone 9.0.B)
- Unit 3 Turbine Building, Basement Floor Elevation 469'-6" Condensate/Condensate Booster Pump area (Fire Zone 8.2.1.B)
- Unit 3 Turbine Building, Basement Floor Elevation 495'-0" Control Rod Drive Pump Mezzanine (Fire Zone 8.2.2.B)
- Unit 3 Reactor Building, Elevation 545'-6' (Fire Zone 1.1.1.3)

b. Findings

On October 29, 2002, while performing a fire protection walkdown of the U2 West LPCI room, the inspector noted degraded insulation around penetration No. F-42-04 (Transco type M2) for an abandoned heating steam line. The inspector reported this condition to the licensee and questioned whether the fire seal was degraded. Operations subsequently declared the penetration inoperable.

The next day the inspector was informed that the degraded insulation did not make up the fire seal and that in accordance with a design drawing there should be 8 inches of fire blanket along the length of the pipe starting from the opposite side of the penetration. The opposite side of the penetration was disassembled and less than 2 inches of fire blanket was found in the penetration. CR# 129421 was written and the penetration was repaired per W.O. 504220-01. The licensee then undertook a program to inspect a sample of M2 penetrations and more discrepancies were found (blanket material ranging between 2 and 6 inches). The licensee subsequently expanded the scope of the inspection program. The licensee identified a total population of 89 Transco type M2 penetrations, and 13 of those inspected as of January 9, 2003, had deficiencies in the amount of fire blanket material. Twelve of the 13 were in their original installation configuration. The licensee is also inspecting 35, type M7 and 90, type M13 penetrations. The degraded penetrations will remain an unresolved item pending the NRC's review of the licensee's completed program **(Inadequate Fire Retardant Material in Fire Barrier URI 50-237/02-17-01).**

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors performed a walkdown of flood protection features for the Unit 3 condenser pit and condensate pump room and the Unit 2 and Unit 3 containment cooling service water pump vaults. The vaults contain safety-related mitigating systems susceptible to flooding from internal sources. During the walkdown the inspectors verified equipment below the flood line was sealed; no holes or unsealed penetrations in floors and walls existed between flood areas; watertight doors between flood areas were maintained and in good material condition; and common drain systems and sumps, including floor drain piping and check valves were operable where credited for flood area isolation.

The inspectors reviewed the corrective actions program database for past flooding events and documentation of previous NRC findings associated with flood protection. The inspectors verified that the licensee entered problems into their corrective action program and the problems were properly addressed for resolution.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07B and A)

a. Inspection Scope

The inspectors reviewed the documents associated with maintenance and thermal performance testing of risk significant heat exchangers. The inspectors reviewed completed surveillance tests, maintenance activities, and associated calculations to confirm that these heat exchangers met their design heat removal requirements or that licensee maintenance practices were adequate to assure design performance.

The inspectors reviewed condition reports concerning heat exchanger or heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues. The inspectors also evaluated the effectiveness of the corrective actions to the identified issues, including the engineering justification for operability, if applicable. The following equipment was evaluated:

- 3A Low Pressure Coolant Injection System Heat Exchanger
- Unit 3 Isolation Condenser Testing and Maintenance

The inspectors also conducted the biennial review of heat sink performance on the Unit 3 isolation condenser. The inspectors reviewed documents associated with testing, inspection, cleaning, and performance trending of heat exchangers primarily focusing on the Unit 3 isolation condenser. This heat exchanger was chosen based upon its importance in supporting required safety functions as well as a relatively high risk achievement worth in the plant specific risk assessment. This heat exchanger was also selected to evaluate the licensee's thermal performance testing methods. During the inspection, the inspectors reviewed a completed surveillance test and associated calculations, and performed independent calculations to verify that these activities adequately ensured proper heat transfer. The inspectors reviewed the documentation to confirm that the test or inspection methodology was consistent with accepted industry and scientific practices, based on review of heat transfer texts and electrical power research institute standards (EPRI NP-7552, Heat Exchanger Performance Monitoring Guidelines, December 1991 and EPRI TR-107397, Service Water Heat Exchanger Testing Guidelines, March 1998) and Mark’s Engineering Handbook.

The inspectors reviewed condition reports concerning heat exchanger and ultimate heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and entering them in the corrective action program. The inspectors also evaluated the effectiveness of the corrective actions for identified issues, including the engineering justification for operability, if applicable.

The documents that were reviewed are included at the end of the report. Also attached is the information request sent to the licensee in preparation for this Heat Sink Inspection.

b. Findings

Introduction

The inspector identified an unresolved item involving inadequate corrective action following a previous Non-Cited Violation in that the licensee again failed to correctly evaluate the test data from performance testing of the Unit 3 isolation condenser. The finding is greater than minor but is unresolved pending completion of the licensee's testing of the Unit 3 isolation condenser and review of a revised analysis planned by the licensee.

Description

On February 6, 2001, the NRC issued Non-Cited Violation (NCV 50-237/01-06-01; NCV 50-249/01-06-01) regarding 10 CFR Part 50, Appendix B, Criterion XI, “Test Control.” Specifically, the licensee had failed to appropriately evaluate test data associated with Unit 3 isolation condenser performance testing. Because the Unit 3 system adequately performed its function following a 1999 scram event and based on the licensee’s engineering judgement, the inspectors believed the system remained operable despite the testing deficiencies. Due to the very low safety significance (Green) of the item and because the licensee entered this item into the corrective action program (CR D2001-00451) on January 24, 2001, the violation was considered a Non-Cited Violation. One of the test evaluation deficiencies was that the licensee failed to properly consider the eductor effect of the reactor recirculation pump running during the test causing a non-conservative overestimation of the isolation condenser’s performance during design conditions. The licensee considered this deficiency appropriately addressed by calculation DRE 02-0020, Revision 0 which was completed May 28, 2002.

The inspector identified the following deficiencies with calculation DRE 02-0020, which were nonconservative, applicable to both units, and repetitive of the previous deficiency:

(1) The licensee failed to properly calculate the eductor effect of the reactor recirculation pump running during the test, because they failed to recognize that some of the pressure energy of a fluid element in the suction pipe would be converted to kinetic energy when the pump was running.

(2) The licensee failed to realize there were design conditions requiring the isolation condenser safety function in which no reactor recirculation pumps would be running, (e.g., after a complete loss of offsite power).

This again caused a nonconservative overestimation of the isolation condenser’s performance during design conditions.

The inspector also identified the following additional concerns with calculation DRE 02-0020:

(3) During the test, the licensee measured the condenser tube mass flow rate to be 322,000 lbm/hr. The licensee attempted to measure the tube side outlet

temperature with a heat gun which indicated impossible temperatures, because they were lower than the shell side fluid temperature of 233 degrees. In the calculation, the licensee assumed that the tube side outlet temperature was 233 degrees Fahrenheit without any reasonable basis. The heat gun was obviously indicating low, but there was no determination of why it was reading low and no determination of how much it was reading low. The inspector compared the test results analysis (based on that assumption) with the isolation condenser specification sheet. The inspector questioned the licensee's temperature assumption because under that assumption the testing data would indicate rather unbelievable results including the following:

- The isolation condenser during the test had about a 23 percent greater heat removal rate than the specification sheet heat removal rate with only about 77 percent of the tubeside mass flow rate under less favorable shell side conditions.

- The isolation condenser would operate during a test at about 160 percent of the specification sheet capacity under less favorable shell side conditions.

- The steam flow into the tubes would be completely condensed and over 300 degrees of subcooling of the liquid would be obtained. However, the inspector's preliminary calculations indicated that complete condensation would require about 70 percent of the tubes' lengths and the 300 degrees of subcooling would require over 70 percent of the tubes' lengths.

(4) The licensee's assumption that the steam entering the isolation condenser after a complete loss of offsite power would be of a high quality similar to the quality during the test at 70 percent power was questionable.

(5) The licensee may not have correctly determined other eductor effects such as that from the jet pumps.

(6) The licensee may not have properly considered the potential effect of the steam velocity during the test increasing the steam flow through the isolation condenser.

(7) The inventory design requirement may not have been properly assessed. Specifically, the inspector was concerned because there was no analysis for the thermal shock of 100 degree water (or less) being injected into the isolation condenser and contacting the tubes that would be at least 540 degrees Fahrenheit as was concluded would happen in Section 7.0 of the calculation.

As a result of these concerns, the licensee planned to provide a revised analysis to the inspector. In addition, the licensee planned additional testing with a revised methodology.

Analysis

The inspector determined that the first two analysis deficiencies constituted an inadequate corrective action and represented a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening.” In particular, the inspector compared this finding to the findings identified in Appendix E, “Examples of Minor Issues,” of IMC 0612 to determine whether the finding was minor. Following that review, the inspector concluded the guidance in Appendix E was not applicable for the specific finding since no examples were provided which involved a failure to properly evaluate these type testing deficiencies. As a result, the inspector compared this performance deficiency to the minor questions contained in Section C, “Minor Questions,” to Appendix B of IMC 0612. The inspector concluded that the issue was more than minor because the finding, if left uncorrected, could become a more significant safety concern. Specifically, the testing deficiencies could allow, as acceptable, an isolation condenser that actually had degraded below its design requirements.

As a result, the inspector reviewed this issue in accordance with IMC 0609, “Significance Determination Process (SDP).” The inspector conducted this review utilizing the “SDP Phase 1 Screening Worksheet for IE [Initiating Events], MS [Mitigating System], and B [Barrier Integrity] Cornerstones.” The inspector determined that this issue potentially affected the NRC’s Mitigating Systems Cornerstone of ensuring the availability of systems that respond to initiating events such as loss of offsite power. However, the inspector could not evaluate the final significance of this issue until the licensee completed planned additional testing with a revised methodology as well as completed a revised analysis of the testing deficiencies to determine if there was any possible impact on isolation condenser operability.

Enforcement

10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” required, in part, that in the case of significant conditions adverse to quality, measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition. Contrary to the above, following an NRC Non-Cited Violation on February 6, 2001, for a failure to appropriately evaluate test data associated with Unit 3 isolation condenser performance testing, a safety-significant condition adverse to quality, the licensee did not adequately determine the cause or take appropriate corrective actions to preclude repetition. Specifically, similar to the original Non-Cited Violation, the licensee’s subsequent calculation DRE 02-0020, Revision 0, completed on May 28, 2002, did not properly consider the eductor effect of the reactor recirculation pump running during the test, causing a nonconservative overestimation of the isolation condenser’s performance during design conditions.

Because the risk significance of this finding is not yet known and because the inspector identified additional concerns regarding calculation DRE 02-0020, Revision 0, that warrant further evaluation, it is considered an unresolved item

(URI 50-237/249/02-17-02) pending planned additional testing with a revised methodology as well as a revised analysis of the testing deficiencies by the licensee.

The licensee entered the corrective action issue into its corrective action program as condition reports 134241 and 134640.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspectors conducted a review of the licensee's inservice inspection (ISI) program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries. Specifically, the inspectors conducted a record review of the following examinations:

WELD #	SYSTEM	Nondestructive Testing TYPE
N2K-1	RPV Shell	Ultrasonic
N2H-1	RPV Shell	Ultrasonic
R8-5831-2	CRD	Liquid Penetrant
14-27A-5	SDC	Ultrasonic

These examinations were evaluated for compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. The inspectors also reviewed inservice inspection procedures, personnel certifications, and NIS-2 forms for Code repairs performed during the last Unit 2 outage to confirm that ASME Code requirements were met.

The inspectors also reviewed a sample of inservice inspection related problems documented in the licensee's corrective action program, to assess conformance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. In addition, the inspectors determined that operating experience was correctly assessed for applicability by the ISI group.

b. Findings

'A' Loop of Reactor Recirculation Flow Sensing Line Socket Weld Failure

On October 8, 2002, the licensee identified that during initial unit 3 drywell entry for the seventeenth refueling outage, the low pressure leg of flow element 3-261-9A, Penetration AR1 on Drawing 382 ('A' Loop Reactor Recirculation), had an unisolable primary coolant leak. The leak was from a 1-inch socket weld. Following repair of the unisolable leak and startup of the unit, the operators noted the return of increased unidentified leakage in the drywell. On December 7, 2002, during a drywell entry to investigate the cause of the return of increased leakage into the drywell, the licensee again identified leakage from the 1-inch socket weld. The licensee subsequently repaired the leak.

The licensee planned to conduct a root cause investigation for both failures. This issue was documented in CR 126277. This issue will be an Unresolved Item (URI) pending the inspectors' review of the licensee's completed root cause investigation **(URI 50-249/02-17-03).**

1R11 Licensed Operator Requalification (71111.11B and Q)

.1 Licensee Requalification Examinations

a. Inspection Scope

The inspectors performed a biennial inspection of the licensee's licenced operator requalification training (LORT) program. The inspectors reviewed the annual requalification operating examination material to evaluate general quality, construction, and difficulty level. The operating portion of the examination was inspected during November 19 - 22, 2002. The operating examination material consisted of two dynamic simulator scenarios and eight job performance measures (JPMs). 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 inspectors reviewed the licensee's program and assessed the level of examination material duplication during the current year annual examinations as compared to the previous year's annual examinations. The inspectors also interviewed members of the licensee's management, operations, and training staff and discussed various aspects of the examination development.

b. Findings

No findings of significance were identified.

.2 Licensee Administration of Requalification Examinations

a. Inspection Scope

The inspectors observed the administration of the requalification operating test to one operating crew to assess the licensee's effectiveness in conducting the test and to assess the facility evaluators' ability to determine adequate performance using objective and measurable performance standards. The inspectors evaluated the performance of one operating shift crew (ten licensed operators divided into two simulator crews) in parallel with the facility evaluators during administration of four dynamic simulator scenarios and five JPMs. The inspectors observed the training staff personnel administer the operating test, including pre-examination briefings, observations of operator performance, individual and crew evaluations after dynamic scenarios, and techniques for JPM cuing. The inspectors evaluated the ability of the simulator to support the examinations. A specific evaluation of simulator performance was conducted and documented under Section 1R11.3, "Conformance With Simulator Requirements Specified in 10 CFR 55.46," of this report. The inspectors also reviewed the licensee's overall examination security program.

b. Findings

No findings of significance were identified.

.3 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, and reactor core performance tests), simulator work request (SWR) records, 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. This evaluation was accomplished by a review of discrepancies noted during the inspection to ensure that they were entered into the licensee's corrective action system and by an evaluation to verify that the licensee adequately captured simulator problems and that corrective actions were performed and completed in a timely fashion commensurate with the safety significance of the item (prioritization scheme). A sample of closed and open simulator discrepancies were reviewed for importance relative to impact on 10 CFR 55.45 and 59 operator actions as well as nuclear and thermal hydraulic operating characteristics.

The inspectors also reviewed the licensee's recent simulator core modeling performance testing to assess the adequacy of the simulator to adequately replicate the actual reactor plant core's performance characteristics. Furthermore, the inspectors conducted interviews with members of the licensee's simulator configuration control group and completed the NRC Inspection Procedure (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).

b. Findings

One Green finding involving a Non-Cited Violation (NCV) of the simulator fidelity regulation, 10 CFR 55.46, was identified for the licensee's failure to adequately maintain simulator fidelity. The inspectors identified two examples of simulator fidelity discrepancies which impacted or would potentially impact operator actions. The finding is greater than minor, and determined to be a Green finding based on Manual Chapter 0609I, "Operator Requalification Human Performance Significance Determination Process (SDP)."

On November 21-22, 2002, the inspectors identified two significant simulator fidelity discrepancies. The discrepancies concerned the adequacy of the licensee to maintain simulator fidelity to demonstrate that control manipulations are completed without procedure exceptions, simulator performance exceptions, or deviation from approved training scenario sequence, in reference to 10 CFR 55.46(c)(2)(ii) as required by 10 CFR 55.46(d)(1), "Continued Assurance of Simulator Fidelity." The discrepancies included the incorrect first stage pressure turbine trip reactor scram bypass setpoint and the incorrect operation of the Reactor Water Cleanup (RWCU) room temperature instrument recorder.

Main Turbine First Stage Pressure Turbine Trip Reactor Scram Bypass Setpoint

On November 21, 2002, the inspectors reviewed the licensee event report (LER) 50-237/2001-005-00, concerning a Unit 2 automatic reactor scram during a reactor startup on November 7, 2001. The LER was of interest due to reasons of lack of operator performance as the causal factor. In addition, the inspectors checked if the licensee incorporated this event as a lessons learned and tested the event on the simulator. The licensee appropriately reviewed the LER event via a simulator test on February 24, 2002. The simulator test revealed that the first stage turbine pressure setpoint was set too high, approximately 275 psig. The correct setpoint was 209 psig. The licensee initiated a corrective action process ticket, SWR-3251, dated March 9, 2002. The identified problem was priority classified as a “Non-Fidelity Issue.” As of November 21, 2002, the inspectors found that SWR-3251 was still active and not yet closed.

After further review, the inspectors found that Unit 2 underwent a power uprate prior to the November 2001 reactor startup. The inspectors also found that Unit 2 implemented a modification on the first stage pressure turbine trip reactor scram bBypass pressure setpoint from 292 psig to 209 psig. This modification was planned as early as January of 2001, as indicated in the engineering evaluation, EC 7984, “Turbine Trip Bypass Setpoint Change.” The inspectors noted that the licensee’s training department was aware of this modification and also of the potential problem associated with the simulator to adequately replicate the actual plant response to first stage pressure. As noted in SWR-1346, “APRM Flow Biased Scram & Rod Block Setpoint,” dated January 4, 2001, the licensee had identified that due to the power uprate, changes may be needed in the setpoints for APRM Flow Biased Scram & Rod Block and Turbine Trip Bypass. The SWR was closed noting that fixed scram setpoints were now installed and the setpoints were also changed to reflect the extended power uprate values. However, the actual plant setpoint change was not finalized until October 4, 2001, in accordance with revision 3 of EC 7984. Apparently the revised EC 7984 was reviewed by training, but an SWR was not updated or issued.

On November 2 and 3, 2001, in preparation for the Unit 2 startup, operators associated with the startup activities were given Just-In-Time (JIT) training. The JIT training was to familiarize the operators of the upcoming plant evolution. However, the simulator was not adequately updated to reflect the new first stage turbine pressure setpoint. The simulator setpoint was set at 275 psig instead of the new 209 psig. The incorrect first stage turbine pressure setpoint potentially affected operator training in a negative way. The apparent inaccurate training did not adequately alert the licensed operators of the potential impact of first stage pressure conditions during an actual reactor startup following the Unit 2 power uprate. The lack of simulator fidelity combined with the operators’ lack of awareness/attention to the plant effects from the turbine first stage pressure led to an actual reactor scram during the November 2001 reactor startup.

Following the reactor scram event, the licensee adequately identified the setpoint discrepancy and continued to track the issue under SWR-3251. However, the inspectors also noted that the licensee had initiated another simulator issue, SWR-3970, “Incorrect setpoint for low power scram bypass,” dated September 6, 2002. This SWR was also priority classified as a “Non-Fidelity Issue.” The inspectors reviewed

SWR-3970 and noted that the licensee had just recently identified that the low power scram bypass appeared to come off of actual reactor power instead of first stage pressure. In addition, SWR-3970 incorrectly noted the first stage pressure setpoint as 290 psig. As of November 21, 2002, the inspectors found that SWR-3970 was still active, with an apparent change made to the first stage pressure to yield 209 psig at a corresponding reactor power of 38.5%.

The inspectors determined that the licensee's training group exhibited lack of follow-up with plant modifications and in maintaining SWRs. Subsequently, the longstanding discrepancy between the plant and the simulator concerning turbine first stage pressure operation and setpoint, led to the inadequate performance of operators which appeared to be a contributing factor to the November 7, 2001, Unit 2 reactor scram.

<u>Reactor Water Clean Up(RWCU) Room Temperature Instrument Recorder</u>

On November 22, 2002, during the administration of simulator scenarios for the annual LORT operating test, the inspectors identified that the RWCU room temperature instrument recorder operated incorrectly. Specifically, the licensee's simulator staff informed the inspectors that with a loss of power to the temperature indication, the recorder was to fail low indicating an indeterminate value. However, the simulator instrument failed in the opposite direction, off-scale high. Subsequently, this simulator discrepancy directly affected operator actions with respect to proper implementation of the Dresden Emergency Plan. The inspectors found that the licensee had apparently identified the simulator discrepancy as SWR-3302, "RWCU room temperature not responding as expected," dated March 18, 2002. The SWR was priority classified only as an "Enhancement." As of November 13, 2002, the inspectors found that SWR-3302 was still active and not yet closed.

The operating test simulator scenario, OPEX-T, included a loss-of-coolant accident inside the drywell along with a loss of Bus 24-1. The expected and correct emergency classification was a Site Area Emergency (SAE) due to the loss of two fission product barriers, Dresden Emergency Action Level (EAL) FS1. The scenario anticipated a General Emergency classification, EAL FG1 (loss of three fission product barriers), only if mitigating actions were ineffective and drywell temperature exceeded 281 degrees Fahrenheit. However, the drywell temperature only increased to approximately 250 degrees Fahrenheit and did not exceed 281 degrees Fahrenheit; therefore, the only correct classification was a SAE. When the Shift Manager reviewed the plant conditions to make his assessment of the emergency classification and protective action recommendations, he additionally referenced the RWCU room temperature instrument recorder. Due to the loss of Bus 24-1, the temperature instrument recorder would have indicated off-scale low or an indeterminate value. However, due to the simulator fidelity discrepancy the temperature instrument recorder indicated the RWCU room temperature as high off-scale exceeding the Max Safe temperature of 210 degrees Fahrenheit.

The Shift Manager incorrectly used this erroneous information to classify the emergency as a General Emergency, in accordance with the Dresden Emergency Plan. The Emergency Plan indicated a potential loss of containment or the third fission product barrier, based on reactor building room temperature exceeding Max Safe condition with an indication of an existing leak in the affected room. The scenario did not have a reactor coolant leak inside the RWCU room. Subsequently, the Shift Manager incorrectly implemented protective action recommendations for offsite evacuation for a General Emergency classification.

The finding, with two examples, was more than minor because the lack of accurate performance of the first stage pressure turbine trip reactor scram bypass setpoint in the simulator apparently led to inaccurate training to the licensed operators. Whereby, the simulator discrepancy failed to adequately alert the licensed operators of the potential impact of the first stage pressure condition following the power uprate during the JIT training for reactor startup activities. The lack of simulator fidelity led to an actual reactor scram during the November 2001 reactor startup due to operators' lack of awareness/attention to the plant effects from the first stage turbine pressure setpoint. In addition, accurate RWCU temperature recorder information was required to ensure the capability to provide accurate temperature assessments and protective action recommendations under accident conditions, as required by the Dresden Emergency Plan. The incorrect operation of the temperature instrument recorder led to an incorrect emergency classification during the recent licensed operator requalification annual operating examination.

Title 10 CFR 55.46(c)(2)(ii) as required by 10 CFR 55.46(d)(1), "Continued Assurance of Simulator Fidelity," states, in part, "Simulator fidelity has been demonstrated so that significant control manipulations are completed without procedural exceptions, simulator performance exceptions, or deviation from the approved training scenario sequence." Contrary to the above, the licensee's failure to effectively maintain continued assurance of simulator fidelity by correcting modeling and hardware discrepancies in a timely manner resulted in a condition which, in combination with operator performance during the November 7, 2001, reactor startup caused an actual reactor scram due to high first stage turbine pressure condition. Although an actual reactor scram occurred due to high first stage turbine pressure, this example of the finding is of very low safety significance because the discrepancy was on the simulator and the actual plant responded as expected to the high first stage pressure and all safety-related equipment functioned properly. In addition, during an approved NRC annual LORT examination simulator scenario the Shift Manager incorrectly classified a Site Area Emergency event as a General Emergency. The incorrect operation of the temperature instrument recorder led to an incorrect emergency classification by the Shift Manager during the recent licensed operator requalification annual operating examination. This example of the finding is also of very low safety significance because the discrepancy was on the simulator and the real recorder in the plant functioned properly. Furthermore, no actual plant emergency occurred and there was no actual impact on equipment or personnel safety.

Because the finding is of very low safety significance and has been entered into the licensee's simulator corrective action process under SWRs-3251, 3302, and 3970, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 50-237/02-17-04 and 50-249/02-17-04, Adequacy of the Plant-Referenced Simulator to Conform With Simulator Requirements Specified in 10 CFR 55.46).**

.4 Biennial Written Examination and Annual Operating Test Results

a. <u>Inspection Scope</u>

The inspectors reviewed the overall pass/fail results of individual written tests, JPM operating tests, and simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee during calender year 2002. The overall results were compared with the significance determination process in accordance with NRC Manual Chapter 0609I, “Operator Requalification Human Performance Significance Determination Process.”

b. <u>Findings</u>

No findings of significance were identified.

.5 <u>Quarterly Observations of Licensed Operator Simulator Training</u> (71111.11Q)

a. <u>Inspection Scope</u>

The inspectors observed Crew #1 on December 6, 2002, during requalification training. The scenario consisted of a small steam leak in drywell, loss of offsite power, and steam cooling. The inspectors also reviewed training examination results of various licensed operators during this period.

b. <u>Findings</u>

No findings of significance were identified.

1R12 <u>Maintenance Rule Implementation</u> (71111.12)

a. <u>Inspection Scope</u>

The inspectors assessed the licensee’s implementation of the maintenance rule by determining if systems were properly scoped within the maintenance rule. The inspectors also assessed the licensee’s characterization of failed structures, systems, and components, and determined whether goal setting and performance monitoring were adequate for the following systems:

- Unit 2 Emergency Diesel Generator
- Unit 2 and 3 High Pressure Coolant Injection
- Unit 2 and 3 Isolation Condenser

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors evaluated the effectiveness of the risk assessments performed before maintenance activities were conducted on structures, systems, and components and verified how the licensee managed the risk. The inspectors evaluated whether the licensee had taken the necessary steps to plan and control emergent work activities. The inspectors used the station's on-line work control process procedure "WC-AA-101" to ensure that the licensee appropriately considered risk factors during the development and execution of planned activities. The inspectors completed evaluations of maintenance activities on the following mitigating systems during this period:

- Replacement of wiring and relays for the failed Unit 2 High Pressure Coolant Injection System signal converter

- Replacement of the cooling coil on Unit High Pressure Coolant Injection System Room Cooler

b. Findings

Unit 3 On-Line Risk Management

A finding of very low safety significance (Green) and an associated NCV of 10 CFR 50.65(a)(4), related to the inadequate assessment and management of risk during maintenance on the high pressure coolant injection (HPCI) system were identified by the inspectors.

On December 16, 2002, the inspectors noted that the licensee had publicized the overall plant risk for Unit 3 as green. During a panel walkdown of the Unit 3 control room main control board, the inspectors noted that the HPCI room cooler and fan were tagged out-of-service. The cooling coil portion of the room cooler was being replaced. The operators considered HPCI system inoperable but available for accident mitigation purposes. The inspectors questioned the Shift Manager on the overall risk assessment of the plant based on actual plant conditions.

The licensee had previously provided the inspectors an analysis that concluded that the HPCI system was able to meet its design basis function without the cooling coil portion of the room cooler being functional. The analysis also concluded that the fan portion of the room cooler needed to be functional to support this analysis. The inspectors reviewed the maintenance schedule for the HPCI system and noted a preventative maintenance activity which involved the removal of the room cooler fan belt. The inspector questioned the licensee on whether this activity had been considered in the decision that the system was available. The licensee responded that this aspect of the work had not been considered in the risk assessment. Additionally, the inspectors identified that before placing the room cooler out-of-service, the licensee had not discussed or preplanned restoring the room cooler fan back to service if the HPCI system received an initiation signal. The licensee's re-evaluation of this issue confirmed the HPCI system was indeed unavailable for this preventative maintenance activity.

The licensee's online work control process as governed by administrative procedure WC-AA-101, "Online Work Control Process," Revision 6, used four levels of risk assessment ranging from lowest to highest risk with associated colors of green (lowest), yellow, orange, and red (highest). The licensee used the ORAM/SENTINEL program to determine the associated risk color.

The inspectors requested the licensee to perform the ORAM/SENTINEL overall risk assessment with the Unit 3 HPCI and the Unit 2 emergency diesel generator systems unavailable since that was the actual plant condition. The Unit 2 emergency diesel generator was considered unavailable for a planned surveillance test on its associated cooling water pump.

When the licensee made the systems unavailable in the ORAM/SENTINEL program, the overall risk changed to yellow with a Core Damage Frequency of 1.55E-05 based on the Plant Transient Assessment Tree. The inspectors determined that although the licensee's risk assessment was not adequate, the Unit 3 isolation condenser system was available and operating as designed; therefore, this finding was determined to be of very low safety significance.

The inspectors reviewed this issue against the guidance contained in Appendix B, "Issue Dispositioning Screening," of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." The inspectors concluded that the issue was more than minor since the finding involved a change in risk level from green to yellow and, if left uncorrected, could become a more significant safety concern. This conclusion was based on the fact that an adequate assessment of risk could have led to additional management strategies including establishment of protected pathways for redundant mitigating systems. The loss of the Unit 3 isolation condenser would have resulted in the plant being in a 'Red' online risk configuration.

The inspectors reviewed this issue in accordance with Manual Chapter 0609, "Significance Determination Process (SDP)," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this Maintenance Rule (a)(4) finding was not addressed in the SDP worksheets and required a Phase 3 evaluation. The Regional Senior Reactor Analyst (SRA) performed a Phase 3 risk assessment and determined that the incremental core damage probability (ICDP) for having the Unit 3 HPCI system unavailable was below the 1E-6 ICDP threshold referenced in NUMARC 93-01, Section 11 (endorsed in NRC Regulatory Guide 1.182) and therefore was of very low risk significance (green) primarily because of the somewhat short duration (1.5 days) that the fan was unavailable.

Title 10 CFR 50.65 (a)(4) requires, in part, that before performing maintenance activities (including but not limited to surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activity. Contrary to the above, the licensee failed to perform an adequate risk assessment when the Unit 3 HPCI system was made inoperable on December 16, 2002. The failure to perform an adequate risk assessment resulted in the licensee inappropriately assigning an overall green risk condition for the plant when actual plant conditions, the unavailability of the HPCI system, warranted a yellow risk assessment. Specifically, the licensee's assessment did not take into account that the room cooler fan would be disabled, making the HPCI system unavailable. The failure to properly perform an adequate risk assessment when the HPCI system was unavailable was an example where the requirements of 10 CFR 50.65 (a)(4), were not met and was a violation. However, because of its low safety significance and because it was entered into the corrective action program as CR 136019 and CR 137916, the NRC is treating this issue as a NCV **(NCV 50-249/02-017-05)**, consistent with Section VI.A.1 of the NRC's Enforcement Policy.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

Throughout the inspection period, the inspectors reviewed operability evaluations (OE) to ensure that operability was properly justified and that the affected component or system remained available, such that no unrecognized increase in risk occurred. The inspectors used the Dresden Updated Final Safety Analysis Report (UFSAR) in assessing the following issues involving system operability:

- Unit 2/3 containment cooling service water & diesel generator cooling water contain non-conforming cast iron components OE 02-010
- High pressure coolant injection room cooler tube leak OE 02-008

b. Findings

No findings of significance were identified

1R16 Operator Work-Around (71111.16)

a. Inspection Scope

The inspectors reviewed operator work-around #02-03-01, "Electro Hydraulic Control Pressure Regulator Failed" to assess any potential effect on the functionality of mitigating systems. During this review the inspectors determined if the operators' ability to implement abnormal or emergency operating procedures was impacted.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modification (71111.17)

a. Inspection Scope

The inspectors reviewed one permanent plant modification associated with torus piping to verify the design adequacy to ensure licensing bases and design bases were maintained, and to ensure functionality of interfacing structures, systems, and components.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance test results to confirm that the tests were adequate for the scope of the maintenance completed and that the test data met the acceptance criteria. The inspectors also reviewed the tests to determine if the systems were restored to the operational readiness status consistent with the design and licensing basis documents. The inspectors reviewed post-maintenance testing activities involving risk significant equipment in mitigating systems and barrier integrity cornerstones:

- 'B' control room heat exchanger heating, air conditioning and ventilation gasket replacement;
- Unit 2 core spray suction valve planned maintenance;
- Unit 2/3 standby gas treatment system planned maintenance.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed and evaluated outage activities performed from October 8-27, 2002, on Unit 3 for a refueling outage and for a subsequent forced outage on Unit 3 which occurred from December 7-11, 2002, due to a repeat leak on a socket weld from the 'A' reactor recirculation sensing line that had been repaired in the recently completed Unit 3 refueling outage. The evaluation was performed to ensure that the licensee appropriately considered risk factors during the development and execution of planned activities. The inspectors conducted walkdowns of systems vital to maintaining the unit in a safe/shutdown condition. The inspectors performed torus and drywell closeout inspections. The inspectors also ensured that Technical Specifications requirements were verified to have been met for changing modes and observed subsequent startup activities.

b. Findings

No findings of significance were identified.

1R23 Temporary Modification (71111.23)

a. Inspection Scope

The inspectors screened active temporary modifications on systems ranked high in risk and assessed the effect of the temporary modifications on safety-related systems. The inspectors also determined if the installations were consistent with system design. The inspectors reviewed the following temporary modifications:

- Engineering evaluation (EC)# 338350 “ Install Temporary Repair to the 2D CCSW Pump Discharge Due to Pin Hole Leak;”
- EC# 339494 “Unit 2 and 3 Turbo Air Inlet Box Leak.”

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed station personnel during an emergency preparedness drill on December 12, 2002, which consisted of a potential bomb threat in the cribhouse, to determine the effectiveness of drill participants and the adequacy of the licensee’s critique in properly determining the emergency classification and identifying weaknesses and failures. The inspectors also observed licensed operator requalification training to determine if proper emergency classifications were made. The training scenario consisted of control rod drift, localized flooding in the condensate booster pump area, loss of all high pressure feedwater, anticipated transient without scram, and emergency depressurization.

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 Plant Walkdowns and Radiological Boundary Verifications

a. Inspection Scope

The inspectors conducted walkdowns of the radiologically protected area to verify the adequacy of radiological area boundaries and postings. Specifically, the inspectors walked down numerous radiologically significant work area boundaries (high and locked high radiation areas) in the unit 2 and 3 reactor buildings (including the unit 3 drywell), the turbine buildings, and the radwaste building and performed confirmatory radiation measurements to determine if these areas and selected radiation areas were properly posted and controlled in accordance with 10 CFR Part 20, licensee procedures, and Technical Specifications. The inspectors also evaluated the radiological condition of those areas walked down to assess the radiological housekeeping and contamination controls.

b. Findings

No findings of significance were identified.

.2 High Radiation Area and Very High Radiation Area Access Controls

a. Inspection Scope

The inspectors reviewed the licensee's procedures, practices and associated documentation for the control of access to radiologically significant areas (high, locked high, and very high radiation areas) and assessed compliance with Technical Specifications, procedures and the requirements of 10 CFR 20.1601 and 20.1602. In particular, the inspectors reviewed the licensee's practices and records for the control of keys to locked high radiation areas (LHRAs) and very high radiation areas (VHRAs), the use of access control guards to control entry into such areas, and the licensee's methods for independently verifying proper closure and latching of LHRA and VHRA doors upon area egress. The inspectors also observed and evaluated the adequacy of the LHRA controls implemented for access into areas of the unit 3 drywell. Additionally, radiological postings were reviewed, and access control boundaries were challenged by the inspectors throughout the plant to verify that high, locked high and very high radiation areas were properly controlled.

b. Findings

No findings of significance were identified.

.3 Review of Radiologically Significant Work

a. Inspection Scope

The inspectors reviewed radiation work permits (RWPs) and as-low-as-is-reasonably-achievable (ALARA) plan packages for selected work activities performed during the Fall 2002 unit-3 refueling outage (D3R17). The inspectors attended pre-job ALARA briefings for the staging equipment in the unit 3 drywell and diving activities to modify the unit 3 reactor steam dryer. The inspectors observed the staging of equipment in the drywell as well as observed selected work activities within the drywell and the unit 3 reactor building. The inspectors also observed activities on the refuel floor to prepare for diving activities in the dryer pool. These activities were performed to verify the adequacy of surveys, access controls, and postings; to assess the exchange of work area radiological information; and to evaluate radiation worker and radiation protection technician performance. The inspectors also evaluated the licensee's procedure and practices for dosimetry placement and use of multiple dosimetry in high radiation areas having significant dose gradients for compliance with the requirements of 10 CFR 20.1201 and applicable regulatory guides. The inspectors compared the requirements for placement of multiple dosimeters to the plan for placing multiple dosimeters on divers during modifications to the unit 3 reactor steam dryer. The licensee's dose tracking and documentation practices were reviewed for work that involved the issuance of multiple whole body and/or extremity dosimetry to verify that worker dose was recorded consistent with 10 CFR 20.2106.

b. Findings

No findings of significance were identified.

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

.1 Radiation Dose Goals and Trending

a. Inspection Scope

The inspectors reviewed job specific and cumulative exposure performance for D3R17 to assess the licensee's dose performance compared to pre-outage exposure goals and projections. The inspectors also reviewed the licensee's dose forecasting practices for radiologically significant jobs completed during the outage to determine if adequate technical bases for job dose estimates existed and to determine if prior outage experiences, resource estimates and industry operating experiences were used to establish reasonable dose estimates. Additionally, the inspectors reviewed the effectiveness of the radiation protection (RP) organization's exposure tracking for the outage to verify that the licensee could identify problems with its exposure performance and take actions to address identified deficiencies.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors reviewed the licensee's procedure for ALARA Plan development. The inspectors evaluated selected D3R17 outage ALARA plans to verify consistency with the procedure and to assess the overall adequacy of the plans relative to both licensee and industry practices. Specifically, the inspectors reviewed the ALARA plans developed for diving activities on the refuel floor, work in the drywell and other high collective dose work activities throughout the unit 3 reactor and turbine buildings to assess the adequacy of the radiological planning associated with each work activity.

The inspectors reviewed the RWP and the ALARA plans completed for each job and assessed the radiological engineering controls and other dose mitigation techniques to verify that they included appropriate controls to reduce dose. These documents were also reviewed to determine if job history files, licensee lessons learned, and industry operating experiences were adequately integrated into each work package. Additionally, the inspectors discussed ALARA planning with involved RP staff to verify that adequate interface between contractors, station work groups, and ALARA staff occurred during job planning.

The inspectors reviewed the exposure results for the selected activities during D3R17 and selected ALARA post - job reviews to evaluate the accuracy of exposure estimates in the ALARA plans. The inspectors compared the actual exposure results versus the initial exposure estimates, the estimated and actual dose rates as well as the estimated and actual man-hours expended. The inspectors reviewed the exposure history for each activity to determine if management had monitored the exposure status of each activity, to determine if work- in-progress ALARA job reviews were needed and had been performed in a timely manner, if additional engineering/dose controls had been established, and if required corrective documents had been generated.

b. Findings

No findings of significance were identified.

.3 Implementation of ALARA Controls and Radiological Oversight of Work

a. Inspection Scope

The inspectors evaluated the execution of the ALARA plans for work activities performed during D3R17 on the refuel floor, within the drywell and in the reactor and turbine buildings. Those activities included the staging of equipment in the unit 3 drywell and on the refueling floor, main steam isolation valve (MSIV) work within the drywell, diving activities to modify the unit 3 reactor steam dryer and maintenance on the unit 3 turbine generator. The inspectors reviewed the adequacy of radiological surveys performed for these jobs, evaluated the radiological work controls, and assessed worker performance and RP staff oversight. Total effective dose equivalent (TEDE) ALARA evaluations were also assessed for technical adequacy. The inspectors evaluated the licensee's radiological engineering controls utilized at these work locations to determine if the controls were consistent with those specified in the ALARA plans. The inspectors also observed and questioned both the RP staff that provided job coverage for these activities and the radiation workers (radworkers) involved in selected work to verify that they had adequate knowledge of radiological work conditions and ALARA controls.

b. Findings

No findings of significance were identified.

.4 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed radiation protection department focus area self-assessment reports on D3R17 outage readiness and preparation, and access control to radiological significant areas and ALARA planning and controls to evaluate the effectiveness of the self-assessment process to identify, characterize, and prioritize problems

The inspectors reviewed the licensee's condition report (CR) database and several individual CRs related to the radiological access control and ALARA programs that were generated during 2002. The review was conducted to assess the effectiveness of the corrective action program to identify problems and to develop corrective actions. Selected CRs were discussed with RP staff and management to determine if problem characterization was accurate and to verify that extent of condition reviews were adequately completed or were in the process of being performed. The inspector also discussed with radiation protection management its practice of conducting both root cause and apparent cause evaluations to determine if they were initiated at appropriate thresholds.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP4 Security Plan Changes (71130.04)

a. Inspection Scope

The inspector reviewed Revision 67 to the Dresden Nuclear Power Station Security Plan to verify that the changes did not decrease the effectiveness of the security plan. The referenced revision was submitted in accordance with regulatory requirements by the licensee letter dated April 11, 2002.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

The inspectors reviewed a sample of plant records and data against the reported performance indicators in order to determine the accuracy of the indicators.

- Unit 2 and Unit 3 Safety System Unavailability, High Pressure Coolant Injection System (October 2001 through October 2002)
- Unit 2 and Unit 3 Safety System Unavailability, Residual Heat Removal System (October 2001 through October 2002)
- Unit 2 and Unit 3 Unplanned Reactor Coolant System Leak Rate (October 2001 through October 2002)

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

The inspectors conducted an inspection of the licensee's corrective action program. The inspectors selected corrective actions for two issues for periodic review of the problem identification and resolution program per NRC inspection procedure (IP) 71152. Additionally, the inspectors verified that: 1) the licensee identified issues at an appropriate threshold; 2) that these issues were correctly entered in the corrective action program; and 3) that these issues were properly addressed for resolution.

b. Findings

Timely Resolution of Issues

.1 Unit 2/3 Emergency Diesel Generator Fuel Oil Transfer Pump

On October 29, 2002, following a surveillance test of the Unit 2/3 emergency diesel generator, operators identified that the Unit 2 power supply feed breaker to the Unit 2/3 emergency diesel generator fuel oil transfer pump had tripped. The licensee entered the appropriate technical specification for this condition, subsequently reset the breaker and, documented this issue in a condition report. The condition report was assigned a 15-day action item to troubleshoot and repair the deficiency. The work priority was changed and the due date was extended to mid-March 2003 through the licensee's work control process. This action item was extended without knowledge of what caused the breaker to trip. On December 3, 2002, operations staff identified that the breaker had tripped again following surveillance testing. The licensee reset the breaker. Unit 2 is the primary power source for the Unit 2/3 emergency diesel generator fuel oil transfer pump. The failure to immediately evaluate the deficient condition led to a delay in the licensee's assessment of determining whether a common mode failure existed with the Unit 3 power supply to the 2/3 emergency diesel generator fuel oil transfer pump.

.2 Fire Hazard in the Unit 2 Turbine Trackway

On November 2, 2002, the licensee identified that a cart, filled with various types of compressed gas cylinders, was stored outside the Unit 2 emergency diesel generator room in the Unit 2 turbine building trackway. The cart contained approximately twelve cylinders of acetylene, oxygen and argon gases. The fire hazard permit attached to the cart documented the storage limit of six oxygen cylinders. Additionally, the licensee's guideline regarding proper storage of combustible gas cylinders states that "Oxygen cylinders in storage shall be separated from fuel gas cylinders or combustible materials." A worker generated a CR and immediately notified the operations staff of the deficient condition. This condition remained unaddressed by the licensee until November 6, 2002, until prompted by the inspectors. The licensee's failure to immediately correct this condition resulted in a known adverse and unsafe condition remaining uncorrected.

4OA3 Event Follow-up (71153)

a. Inspection Scope

The inspectors reviewed licensee event reports (LERs) to ensure that issues documented in these reports were adequately addressed in the licensee's corrective action program. The inspectors also reviewed an unresolved item to determine if the licensee was in non-compliance with any regulatory requirement. The inspectors also interviewed plant personnel and reviewed operating and maintenance procedures to ensure that generic issues were captured appropriately.

The inspectors reviewed operator logs, the Updated Final Safety Analysis Report, and other documents to verify the statements contained in the LER.

b. Findings

.1 (Closed) LER 50-237/2002-005-00: "Pressure Switches Found Above Technical Specification Allowable Values"

On June 4, 2002, while performing DIS 0250-03, Revision 34, "Electromatic Relief Valve/Target Rock Valve Pressure Switches Calibration Without Control Switch Functional Testing," the instrument maintenance department personnel found that the pressure switch setpoint for Target Rock Valve 2-203-3A (as-found at 1134.6 psig) exceeded the Technical Specification (TS) Allowable Value of $\leq$1133.5 psig, and the pressure switch setpoint for Electromatic Relief Valve 2-203-3B (as-found at 1110.6 psig) exceeded the TS Allowable Value of $\leq$1110.5. The licensee subsequently readjusted each pressure switch to within procedural tolerances and declared each switch operable prior to proceeding to the next switch.

The test pressure, as specified by DIS 0250-03, was required to be reduced to zero after as-found data was collected and before the pressure switch was calibrated. After contacting the pressure switch vendor, the licensee determined that the practice of reducing the test pressure to zero promoted instrument drift. The licensee subsequently revised three procedures, including DIS 0250-03, to maintain operating system pressure at the pressure switch, at all times, during testing except when increasing or decreasing pressure toward the setpoint.

The safety significance of this event was minimal because (1) the automatic depressurization system function and manual open function of the valves remained operable; (2) the safety relief mode of the Target Rock valve was not affected; (3) the as-found condition of the two pressure switches would have resulted in a slight delay in the opening of their associated reactor pressure vessel relief valves; and (4) the analytical limit for the pressure switches was not exceeded; therefore, the associated relief valves would have maintained reactor vessel pressure within design limits upon a turbine trip without bypass valve capability.

Although this TS violation was entered into the licensee's corrective action program as CR 110632, this issue constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This LER is closed.

.2 <u>(Closed) LER 50-249/2002-004-00</u>, "Main Steam Safety Valves Failed the Technical Specification(TS) As-Found Lift Setpoint"

Four main steam safety valves were removed and tested during the Fall 2002 Unit 3 refueling outage as specified by the testing frequency of the inservice testing program. Two failed the 1% lift setpoint verification as required by TS Surveillance Requirement 3.4.3.1 which required that 2 valves lift at 1240 psig +-12.4 psig, 2 valves lift at 1250 psig +-12.5 psig, and 4 valves lift at 1260 psig +-12.6 psig. One valve lifted at 1208 psig when its nameplate setpoint listed at 1240 psig and the other valve lifted at 1227 psig when its nameplate setpoint listed at 1260 psig. The two valves were within the inservice testing program requirements of lifting within 3%. The licensee determined the root cause of the valves lifting outside the TS limit was setpoint drift. The failure of the valves to lift within the required technical specified limit is a violation.

The safety significance of this event is minimal due to the valves being able to provide overpressure protection with valves lifting below the TS setpoint value. The valves were inspected, refurnished and tested. The licensee is planning to submit a TS amendment request to increase the setpoint tolerance of the valves. Also, the licensee will perform all required analysis to support the increased setpoint tolerance.

All four removed valves were replaced by previously acceptably tested valves to the 1% TS requirement.

Although this TS violation was entered into the licensee's corrective action program as CR 127325, this issue constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This LER is closed.

.3 (Closed) Unresolved Item (URI) 50-249/01-09-02: Review of the Licensee's Resolution to an Inadequate Operability Evaluation for a Degraded Unit 3 Emergency Core Cooling System (ECCS) Suction Strainer.

In June of 2001, inspectors identified an URI associated with an inadequate operability evaluation that the licensee had completed for loose and missing bolts found on the support flanges for the Unit 3 ECCS suction strainers. Specifically, the operability of the most degraded strainer was evaluated in calculation DRE99-0028, "Historical Operability Evaluation For Dresden Unit 3 ECCS Suction Strainer Flange For Loose Bolts Found During D3R15." In this calculation, the licensee concluded that the strainer support flange met operability stress criteria. However, in this calculation, the licensee incorrectly used the moment of inertia for a solid circle to determine flange loading instead of a hollow circle, and incorrectly used a factor of two reduction for the dynamic section modules that was already accounted for in the equivalent dynamic pressure equation (e.g., double counting the reduction factor for dynamic loads). Further, the operability stress criteria applied were reviewed and accepted by the NRC for use on piping or structural supports, not flanges. The licensee entered this finding into the corrective action system (D2001-03072) and subsequently corrected these errors in Revision 1 to DRE99-0028. Based on this revised calculation, the licensee demonstrated that the degraded suction strainer met applicable operability criteria.

The inspectors compared this finding to the findings identified in Appendix E, "Examples of Minor Issues," of IMC 0612 to determine whether the finding was minor. Following that review, the inspectors identified that this issue was similar to Example 3.a of Appendix E, of IMC 0612 related to a technical calculation error that was considered a minor violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The licensee corrected this finding and it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This URI is closed.

4OA5 Other Activities

.1 Completion of Appendix A to TI 2515/148, Rev 1

The inspector completed the pre-inspection audit for interim compensatory measures at nuclear power plants, dated September 13, 2002.

.2 Power Uprate (71004)

a. Inspection Scope

The inspectors reviewed a number of extended power uprate modifications to verify that the modifications were prepared in accordance with the licensing basis and the Updated Final Safety Analysis Report. In addition, the inspectors reviewed the modifications to verify that the mitigating system capability would be maintained. The inspectors also reviewed prepared modifications to ensure that the licensee properly performed design change evaluations consistent with 10 CFR Part 50.59, "Changes, Tests and Experiments." The modifications reviewed included Main Steam and Torus Attached Piping.

b. Findings

No findings of significance were identified.

4OA6 Exit Meetings

.1 The resident inspectors presented their inspection results to Mr. R. Hovey and other members of licensee management at the conclusion of the inspection on January 3, 2003.

The licensee acknowledged the findings presented. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Radiation Protection inspection with Mr. R. Hovey on October 10, 2002.
- Inservice inspection with Mr. R. Hovey on October 22, 2002.
- Safeguards inspection with Ms. V. Gangler on October 30, 2002.
- Licensed Operation Requalification inspection with Mr. R. Hovey on November 25, 2002 (71111.11A).
- Radiation Protection inspection with Mr. M. Phalen on December 10, 2002.
- Heat Sink Inspection with R. Hovey, Site Vice President and D. Bost, Plant Manager on December 10, 2002.
- Licensed Operation Requalification inspection with Mr. J. Lindsey on December 20, 2002 (71111.11B), via telephone.

4OA7 Licensee Identified Violation

The following finding of very low safety significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as a Non-Cited Violation (NCV).

Cornerstone: Initiating Event

NCV Tracking Number 50-237;249/02-017-07	Title 10 CFR 50.55a, 50-237/02-017-07 “Codes and Standards,” paragraph (g)(4)(ii) states that inservice examination of components and systems pressure tests must comply with the requirements of the latest edition and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The 1998 Edition, ASME Boiler and Pressure Vessel Code, Section XI, requires surface examinations of welds in control rod drive (CRD) housing tubes on 10 percent of peripheral drives. These welds are required to be inspected once every ten years. Exelon Procedures ER-AA-330 and ER-AA-330-002 govern the implementation of the ASME Section XI ISI Program. Contrary to the above, on June 17, 2002, as described in the licensee’s CR 113590, during a review of the ISI program in preparation of the fourth ten year inspection interval, the licensee identified that they had not incorporated the CRD housing tubes pipe to pipe welds in the ISI program when the inspection requirements were implemented during the first inspection. The ASME Section XI requirements were not met as required by procedures. This is a violation of 10 CFR 50.55a, “Codes and Standards.” The licensee subsequently added the subject welds to the ISI component population, examined the welds finding no indications, and entered the issue into the corrective action program.

KEY POINTS OF CONTACT

Licensee

R. Bauman, ISI Coordinator
S. Bell, Health Physicist
D. Bost, Station Director
H. Bush, Lead Radiation Protection Supervisor
J. DeYoung, Corporate EP Specialist
J. Ellis, Performance Monitoring Group Lead
T. Fisk, Chemistry Manager
J. Ferguson, ALARA Analyst
R. Gadbois, Shift Operations Superintendent
V. Gengler, Dresden Site Security Director
R. Geier, RV/ISI NDE Coordinator
T. Green, NDE Level III
K. Hall, NDE Level III
J. Hansen, Regulatory Assurance Manager
J. Heaton, Corporate Operations Training Director
J. Henry, Operations Director
R. Hovey, Site Vice President
S. Hunsader, Corporate Maintenance Rule Owner
J. Lindsey, Operations Training Group Lead
T. Loch, Supervisor, Turbine Systems Group
R. May, NDE Level III
C. Melgoza, ALARA Analyst
S. Nelson, Senior System Engineer, Turbine Systems Group
D. Nestle, Radiation Protection
P. O'Connor, Simulator Supervisor
M. Overstreet, Radiation Protection Shift Supervisor
M. Pavey, Emergency Preparedness Coordinator
M. Phelan, Assistant Radiation Protection Manager
J. Reda, Design Engineer
T. Richmond, Learning Services Manager
R. Ruffin, Regulatory Assurance - NRC Coordinator
A. Shahkarami, Engineering Director
J. Sipek, Nuclear Oversight Director
N. Spooner, Site Maintenance Rule Coordinator
B. Svaleson, Maintenance Director
C. Symonds, Training Director
S. Taylor, Radiation Protection Director
D. VanAken, Corporate EP Specialist

NRC

M. Ring, Chief, Division of Reactor Projects, Branch 1
D. Smith, Dresden Senior Resident Inspector
B. Dickson, Dresden Resident Inspector
P. Pelke, Reactor Engineer

M. Sheikh, Nuclear Safety Intern/Fuel Cycle Inspector

IDNS

R. Zuffa, Illinois Department of Nuclear Safety
R. Schulz, Illinois Department of Nuclear Safety

Contractor

J. Easton, Project Manager, General Electric
S. Snyder, ISI Coordinator, General Electric

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-237/02-17-01	URI	Inadequate Fire Retardant Material in Fire Barrier
50-237/02-17-02 50-249/02-17-02	URI	Isolation Condenser Performance Testing Deficiencies
50-249/02-17-03	URI	'A' Loop of Reactor Recirculation Flow Sensing Line Socket Weld Failure
50-237/02-17-04; 50-249/02-17-04	NCV	Adequacy of the Plant-Referenced Simulator to Conform With Simulator Requirements Specified in 10 CFR 55.46
50-237/02-17-05	NCV	Failure to Perform an Adequate Assessment of Risk When High Pressure Coolant Injection System was Unavailable

Closed

50-237/2002-005-00	LER	Pressure Switches Found Above Technical Specification Allowable Values
50-249/2002-004-00	LER	Main Steam Safety Valves Failed the Technical Specification As-Found Lift Setpoint
50-249/01-09-02	URI	Review of the Licensee's Resolution to an Inadequate Operability Evaluation for a Degraded Unit 3 Emergency Core Cooling System (ECCS) Suction Strainer
50-237/02-17-04; 50-249/02-17-04	NCV	Adequacy of the Plant-Referenced Simulator to Conform With Simulator Requirements Specified in 10 CFR 55.46
50-237/02-17-05	NCV	Failure to Perform an Adequate Assessment of Risk When High Pressure Coolant Injection System was Unavailable

Discussed

None

LIST OF ACRONYMS USED

ALARA	As Low As Is Reasonably Achievable
APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
D3R17	Dresden Unit 3 Refueling Outage
DIS	Dresden Instrument Surveillance
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EAL	Emergency Action Level
EC	Engineering Evaluation
ECCS	Emergency Cooling System
EDG	Emergency Diesel Generator
EPRI	Electrical Power Research Institute
HPCI	High Pressure Coolant Injection
ICDP	Incremental Core Damage Probability
IDNS	Illinois Department of Nuclear Safety
IP	Inspection Procedure
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
JIT	Just In Time
JPM	Job Performance Measure
KW	Kilowatts
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LORT	Licensed Operator Requalification Training
MSIV	Main Steam Isolation Valve
MWe	megawatts electrical
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OA	Other Activities
OE	Operability Evaluation
psig	pounds per square inch gauge
Radworker	Radiation Worker
RP	Radiation Protection
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
RWP	Radiation Work Permit
SAE	Site Area Emergency
SDP	Significance Determination Process
SWR	Simulator Work Request
TEDE	Total Effective Dose Equivalent
URI	Unresolved Item
VHRA	Very High Radiation Area
WO	Work Order

LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather Protection

DOS 0010-22	Preparation for Cold Weather for Unit 2	Revision 10
DOS 0010-23	Initiation of Cold Weather Operations for Unit 2	Revision 5
DOS 0010-25	Preparation for Cold Weather for Unit 3	Revision 11
DOS 0010-26	Initiation of Cold Weather Operations for Unit 3	Revision 3
DOS 0010-28	Preparation for Cold Weather for Radwaste	Revision 12
DOS 0010-29	Initiation of Cold Weather Operations for Radwaste	Revision 3
CR 125536	Nuclear oversight winter readiness deficiencies	October 2, 2002

1R04 Equipment Alignment

CR 128856	Plant labeling - a future issue to consider	October 23, 2002
CR 129219	Reactor water clean up demin by pass left open	October 27, 2002
CR 129448	Operations self identifies misuse of “Equipment Status Tag”	October 31, 2002
DOP 2300-1/E1	Unit 2 high pressure coolant injection system	Revision 28
DOP 1400-M2	Emergency core cooling system fill system	Revision 09
DOP 1400-E1	Core spray electrical	Revision 03

1R05 Fire Protection

CR 124695	NEIL recommendations regarding fire protection audit	September 26, 2002
CR 124886	Transient combustibles in trackway 3	September 27, 2002
CR 125175	Operations identifies apparent violation of Dresden OP-AA-201-003	September 30, 2002
CR 125180	Operations identifies apparent lack of configuration control	September 30, 2002
CR 125618	Nuclear Oversight identifies fire wrap inspection by-passed	October 2, 2002
CR 125622	Welding over bottle station	October 2, 2002
CR 125679	Fire hose found pressurized	October 2, 2002
CR 125987	CCA # 123416 identifies common cause	October 4, 2002

CR 126497	WSI firewatch/hot work deficiencies in heater bay	October 9, 2002
CR 126506	Blocked fire equipment	October 8, 2002
CR 126529	Blocked fire door	October 9, 2002
CR 126550	Unplanned TRM entry for fire door #120	October 9, 2002
CR 127280	Safe shutdown light discovered with all lights lit	October 14, 2002
CR 127288	Fire cart not chocked	October 14, 2002
CR 128467	Wood accumulating in unit 2 turbine trackway	October 22, 2002
CR 128636	Poor housekeeping on unit 3 condensate pump floor	October 23, 2002
CR 128794	Scaffold storage in trackway 3	October 24, 2002
CR 128959	Blocked fire extinguisher	October 25, 2002
CR 129392	NRC identifies concerns in the 2/3 emergency diesel generator room	October 28, 2002
CR 129421	NRC Comments from Plant Tour	October 29, 2002
CR 129959	NRC questions gap in fire door 67	October 31, 2002
CR 133752	Identified safe shutdown light #365 electrolyte level low	
CR 132462	Transco M-2 type fire barriers lack 8" of fire blanket	November 15, 2002
DFPS 4175-07	Fire Door/Oil Spill Barrier Maintenance	Revision 15
Dresden Unit 2 Fire Pre-Plan U2RB-7	Unit 2 Reactor Building, Elevation 545'-6"	Fire Zone 1.1.2.3
Dresden Station Units 2 and 3 Updated Fire Analysis Section 4.2.3	Reactor Building, Elevation 545'-6"	Fire Zone 1.1.2.3
Dresden Unit 2 Fire Pre-Plan U2RB-2	Unit 2 Reactor Building Elevation 476'-6" West Low Pressure Coolant Injection Corner Room	Fire Zone 11.2.1
Dresden Station Units 2 and 3 Updated Fire Analysis Section 4.2.7	Low Pressure Coolant Injection Pump Room - Division II	Fire Zone 11.2.1

Dresden Unit 2 Fire Pre-Plan U2RB-4	Unit 2 Reactor Building Elevation 476'-6" High Pressure Coolant Injection Room	Fire Zone 11.2.3
Dresden Station Units 2 and 3 Updated Fire Analysis Section4.7.3	Unit 2 High Pressure Coolant Injection Pump Room	Fire Zone 11.2.3

1R06 Flood Protection

CR 121643	D3 containment cooling service water pump vault penetration seals fail as-found leak test	September 4, 2002
CR 114738	Unit 3 containment cooling service water pump vault penetration seal failures	June 6, 2002
CR 114725	Unit 2 containment cooling service water pump vault penetration seal failures	May 1, 2001
CR 122126	D2 containment cooling service water pump vault penetration seals fail as-found leak test	July 3, 2002
CR 126419	Reactor building floor drain sump hi level alarm	October 8, 2002
CR 128597	3-A equipment drain sump pump failure after cleaning bellows	October 23, 2002
CR 128654	3B drywell emergency diesel sump pump found clogged with debris after removal	October 23, 2002
CR 129435	CCSW valve	
CR 131259	Penetration found with no fire stop	November 12, 2002
CR 131542	Station fire marshall and NRC identified concern	November 12, 2002
CR 131640	Auto start of 2/3 diesel fire pump	November 14, 2002
CR 131664	NRC/IDNS inspectors identify deficiencies during plant tour	November 18, 2002
WO 00495451-01	Replace the high level float switch in the 3B reactor building floor drain sump	October 11, 2002
UFSAR 3.4.1.2	Internal Flood Protection Measures	
DOA 0010-04, Revision 15	Floods	September 10, 2002
AR No. 0011105	The NRC senior resident inspector identified several weaknesses in procedure DOA 0010-04	Revision 14

ER-MW-450	Structures monitoring	Revision 0
Drawing FL-1, Revision A	Flood barriers basement floor	May 26, 1995
Drawing FL-19, Revision A	Flood barrier unit 3 turbine building, sheets 1 and 2	May 26, 1995
Drawing FL-20, Revision A	Flood barrier unit 3 turbine building, sheets 1 and 2	May 26, 1995
Drawing FL-23, Revision A	Flood barrier unit 3 turbine building, sheets 1 and 2	May 26, 1995
Drawing FL-24, Revision A	Flood barrier unit 3 turbine building, sheets 1 and 2	May 26, 1995
Drawing FL-25, Revision A	Flood barrier unit 3 turbine building, sheets 1 and 2	May 26, 1995
Drawing FL-29, Revision A	Flood barrier unit 3 turbine building, sheets 1 and 2	May 26, 1995
Drawing FL-30, Revision A	Flood barrier unit 3 turbine building, sheets 1 and 2	May 26, 1995
Drawing B-440, Revision A	Typical details for sealing floor and wall openings	August 28, 2001
Drawing B-442, Revision K	Sealing Floor and Wall Opening, Sheet 2	August 28, 2001
Drawing 12E-6508, Revision A	Electrical installation air-seals and fire stops	August 28, 2001
WR 980094549-01	D3 6Y PM replace solenoid on containment cooling service water vault drain AO 3-4999-74	
WR 960105706-01	D2 6Y PM replace solenoid on containment cooling service water vault drain AO 2-4999-74	
WR 990096773-01	D3 18M TS containment cooling service water drain valve test	
WR 99149470-01	D2 18M TS containment cooling service water drain valve test	

1R07 Heat Sink Performance

CR 134640	Additional issues associated with ISCO heat performance test	December 6, 2002
Design Analysis DRE02-0020	Isolation Condenser Heat Removal Capacity Test Validation	Revision 0
DOS 1300-01	Isolation Condenser Five Year Heat Removal Capability Test Performed on Unit 3 on January 17, 2002	Revision 19
Job Number 66-2-5636 & 5637	General Electric Struthers Wells Isolation Condensers' Heat Exchanger Specification Sheets	February 15, 1967
CR 2001-0451	NRC Identified Weaknesses Associated with the Isolation Condenser Five Year Performance Testing	January 24, 2001

1R07 Condition Reports Initiated as a Result of Inspection

CR 133213	Self Assessment Findings in Preparation for NRC Heat Sink Inspection	November 25, 2002
CR 134241	The NRC Identified That Calculation DRE 02-0020 Failed to Consider Design Cases Where the Isolation Condenser Would Need to Operate Without the Reactor Recirculation Pump Operating	December 5, 2002
CR 134640	The NRC Identified That Calculation DRE 02-0020 Failed to Recognize That the Pressure in the Suction Pipe of the Reactor Recirculation Pump is Reduced by the Velocity Head if the Pump is Running	

1R08 Inservice Inspection

MT-EXLN-100V3	Procedure for Magnetic Particle Examination	October 2001
PT-EXLN-100V3	Procedure for Liquid Penetrant Examination Using Fluorescent and Visible Liquid Penetrant Inspection Methods	October 2001
GE-UT-311	Procedure for Manual Ultrasonic Examination of Nozzle Inner Radii and Bore	October 21, 2001
GE-PDI-UT-1	PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds	September 25, 2002

	ISI Program Plan, Third Ten-Year Inspection Interval, Dresden Station, Units 2 & 3	September 24, 2002
	Dresden Nuclear Power Station, Unit 2, Inservice Inspection (ISI) Summary Report, Fall 2001 Inservice Inspection Period	February 5, 2002
Relief Request # CR -21	(NRC Letter) Relief for Risk - Informed Inservice Inspection of Piping	September 5, 2001
CR 126887	Ultrasonic Indication Detected on 3B Lower Sensing Line	
CR 127079	Nondestructive Examination Rejectable Indication (Isolation Condenser)	
CR 127897	NDE Ultrasonic Rejectable Defects (Feedwater)	

1R11 Licensed Operator Requalification

DGP 01-01	Unit Startup; Rev. 101	no date
DGP 01-S1	Startup Checklist; Rev. 55	no date
DGP 01-S5	Mode 3 to Mode 2 Restart Checklist; Rev. 05	no date
DGP 02-01	Unit Shutdown; Rev. 70	no date
TQ-AA-106	Licensed Operator Requal Training Program; Rev. 2	no date
TQ-AA-106-0304	Licensed Operator Requal Training Examination Development Job Aid; Rev. 1	no date
TQ-AA-301	Simulator Configuration Management; Rev. 1	no date
TQ-AA-301-0301	Simulator SWR Prioritization Flowchart Job Aid; Rev. 0	no date
TQ-AA-301-0302	Simulator MOD/DCR Prioritization Flowchart Job Aid; Rev. 0	no date
TQ-AA-302	Simulator Certification Testing and Reporting, Rev. 2	no date
TQ-AA-302-0101	Simulator Test Procedure Cover Sheet, Rev. 0	no date
TQ-AA-302-0102	Simulator License Event Report (LER) Test Procedure, Rev. 0	no date
PSLTR: #02-0003	Licensee letter to NRC, Subject: License Event Report 2001-005, “Unit 2 Scram due to Increased First Stage Turbine Pressure”	January 7, 2002

SWR-3970	Simulator Work Request - Incorrect setpoint for low power scram bypass	September 6, 2002
SWR-3302	RWCU Room Temperatures not responding as expected	March 18, 2002
SWR-3251	LER 2-2001-05 (Scram During Turbine Shell Warming)	March 9, 2002
SWR-1346	APRM Flow Biased Scram and RB Setpoint	January 4, 2001
SWR-4035	Cert Transient Testing	September 20, 2002
SWR-3085	Simulation Failed	February 6, 2002
SWR-3293	EDG Issue	March 15, 2002
SWR-618	New Cooling Towers; dated May 10, 1999	April 18, 2002
SWR-3967	Feedwater Heaters tripping too soon	September 6, 2002
SWR-949	Turbine valve response is not correct on a turbine roll; dated January 21, 2000	March 24, 2001
SWR-3663	Loss of 125 Vdc Annunciator Power audible 'ding'	June 7, 2001; updated June 11, 2002
SWR-4173	Station Blackout DCS intermittent spurious alarm	November 1, 2002
SWR-3138	2/3 EDG kilovar meter	May 25, 2001; updated September 30, 2002
EC-7984	Turbine Trip Bypass Setpoint Change; Rev. 3	September 27, 2001
D2C18	Advanced Core Model Test Series - Stand Alone Benchmark Testing	July 26, 2001
DRE-AM-CR-09	Hot Full Power Steady State Condition; Rev. 0	August 10, 2001
DRE-AM-CR-05	Rod Worth Values	August 10, 2001
DRE-AM-CR-03	Moderator Temperature Coefficient of Reactivity; Rev. 0	August 10, 2001
DRE-AM-CR-02	Doppler Coefficient of Reactivity	August 10, 2001

DRE-AM-CR-01	Void Coefficient of Reactivity	August 10, 2001
	Simulator Test - DOS 1400-05; Core Spray System Pump Test with Torus Available	June 15, 2001
	Simulator Test - DOS 0250-01; Partial Closure Operability Test of Main Steam Isolation Valves	June 15, 2001
	Simulator Test - Malfunction T13; DG 2/3 Auto Start Relay 2/3-2 Failure; Rev. 0	February 17, 2001
	Simulator Test - HPAOPASF; HPCI Aux Oil Pump Failure to Auto Start	February 9, 1999
	Simulator Test - Malfunction HP3; Feedwater System Leak	February 10, 2001
	Simulator Test - Malfunction HP4; Feedwater System Leak in Drywell	March 29, 2001
	Simulator Work Request System Help Guide; Rev. 2	no date
	List of all ‘Open’ Simulator Work Request as of August 16, 2002	dated various
	Simulator versus Reference Plant (Unit 2) Differences; Rev. 4	no date
	Dresden Simulator ANSI/ANS-3.5 Certification Report (Report Update)	March 2002
	Dresden 2002 Annual Operating Exam Material - Training Cycle 6-2002; two simulator scenarios; eight JPMs	dated various

1R12 <u>Maintenance Rule Implementation</u>

CR 117766	Received unexpected alarm. High pressure coolant injection turbine inlet drain pot hi (902-3, B-11)	
CR 108239	Unexpected main control room alarm - high pressure coolant injection system	
CR 1298895	Unexpected main control room alarm (902-3 tile B11)	
CR 124779	System Health Indicator Program (SHIP) focus area self-assessment	September 20, 2002

1R13 Maintenance Risk Assessments and Emergent Work Control

WO 514250-02	High pressure coolant injection signal converter failure/replace wiring and relays	

1R15 Operability Evaluations

CR 125281	Use of WD-40 inside of “B” filter unit	October 1, 2002
CR 125383	613' heavy load lift requirements not met per DMP 5800-03	September 23, 2002
CR 125970	125 Vdc battery sizing calculation discrepancies	October 4, 2002
CR 126054	Portable manlift found against motor control center 38-7	September 28, 2002
CR 126277	Primary coolant leak from 3A RR pump discharge header	October 8, 2002
CR 128528	Potential degradation of the main control room ventilation	
CR 128320	NRC notes corrosion in Torus	
CR 131852	Belleville spring missing from 3-0203-1D main steam isolation valve	November 18, 2002
CR 133191	Unexpected high pressure coolant injection signal converter failure alarm	November 26, 2002
CR 134120	2C electromagnetic relief valve indication reading abnormally low	
CR 132384	Two scram valve casing nuts missing on scram valve	November 19, 2002

1R16 Operator Workarounds

CR 126009	Unable to maintain primary containment pressure	October 4, 2002

1R19 Post Maintenance Testing

CR 129140	Technical Specification required post maintenance tests not assigned to CRD work orders	October 25, 2002

1R20 Refueling and Outage

CR 124920	Nuclear oversight observations on 2/3 refueling floor	September 27, 2002
CR 125143	operations identifies the following deficiencies on refuel floor	September 30, 2002

CR 126757	Technical Requirement Manual 3.4.A interpretation - 3A RR pump sensing line weld crack	October 9, 2002
CR 126797	Feed water 3-0220-58A failed as found local leak rate test	October 9, 2002
CR 126801	Main steam line drains 3-0220-1 and 0220-2 failed local leak rate test	October 10, 2002
CR 126803	High pressure coolant injection 45 valve failed as found local leak rate test	October 10, 2002
CR 126965	Leakage discovered on drywell pedestal during Decommissioning Technical Specifications 1600-06	October 10, 2002
CR 127079	Nondestructive examination rejectable indication	October 12, 2002
CR 127092	Foreign material found on core plate	October 12, 2002
CR 127186	New nut found next to jet pump #1	October 14, 2002
CR 127462	Plywood decking found inside reactor support skirt	October 13, 2002
CR 128125	Diver dose attributed to wrong radiological work permits	
CR 128320	NRC notes corrosion in Torus	October 21, 2002
CR 128811	NRC concerns from unit 3 drywell walkdown	October 24, 2002
CR 128973	NRC Drywell closeout inspection items	October 24, 2002
WO#00442270-03	Remove the first layer of shield blocks above the unit 3 reactor vessel	September 4, 2002
DGP 02-01	Unit shutdown	Revision 41
DGP 04-01	Refueling	Revision 01
WO#9660040122	3B RR sensing line Inspection	
WO#9921499001	Recirculation Flow CAL (APRM flow variable) DIS 700-01	
WO#99264220	Replace MSL high flow switches	
WO#992268910	Repair replace 3A low pressure coolant injection cooler (fan only)	
WO#9912605602	Provide seismic support for 3B low pressure coolant injection cooler	

1R22 Surveillance Test

CR 125181	Out of tolerance non technical specification	September 24, 2002
CR 127069	3-1601-55, 1601-31B and 8526RV, failed as found local leak rate test	October 12, 2002
CR 127248	D3R17 performance of DIS 2300-07	October 12, 2002
CR 128941	Failure to perform VT-2 inspection as required	October 25, 2002
CR 128609	Unit 3 drywell vent fans started without performing DTS 5750-01	
CR 128799	Unexpected slow scram times on some control rod drives	
CR 129128	Received unexpected alarms during DIS 2500-03	October 28, 2002
CR 129135	Phantom 903-55 panel annunicator A-4	October 28, 2002
CR 129371	Operations peer check expectations of work groups	October 29, 2002

1R23 Temporary Plant Modifications

CR 125991	250 Vdc motor control center breaker modification deficiencies discovered	October 3, 2002
CR 127174	Relays in the 2203-73A panel not installed prior to testing	October 13, 2002
CR 129079	Temporary modification EC not installed per installation drawings	October 27, 2002

71152 Problem Identification and Resolution

CR 124144	Inadequate Mazon Inprocessing training	September 23, 2002
CR 125084	DC System	September 27, 2002
CR 125161	Ineffective corrective actions for CR 103935	September 30, 2002
CR 125242	Operations team 2 crew clock due to condition report initiation threshold	September 30, 2002
CR 125587	Condition report not processed timely	October 2, 2002
CR 126159	Condition report not processed timely	October 7, 2002
CR 127209	Operability determination 02-013 corrective action extended	October 14, 2002
CR 132183	Work orders supporting corrective actions being canceled	November 21, 2002

CR 134239	Unit 3 reactor building vent modification	December 10, 2002

71153 Event Follow-up

CR 127325	Two main steam safety valves failed the technical specification 1% lift setpoint	October 20, 2002
CR 110632	2-203-3A and 2-203-3B pressure switches out-of-tolerance	
DIS 0250-03	Electromatic Relief Valve/Target Rock Valve Pressure Switches Calibration Without Control Switch Functional Testing	Revisions 34, 35, and 36
CR 116478	Unit 3 reactor scram due to turbine trip	
CR 115691	Loss of main turbine speed indication/PMG failure	

71004 Power Uprate

CR 127036	Deficiencies in Bailey RFP runout logic modification, WO 99248774-04	October 12, 2002
CR 128907	Calculation error in extended power uprate rev. calc. # 29.0201.0211-35	October 25, 2002
CR 128661	Maximum combined flow limiter settings	October 22, 2002
CR 130067	Revision of feed water level control extended power uprate tuning procedure, SP 02-07-008	November 2, 2002

2OS1 Access Control to Radiologically Significant Areas

DRS 5600-01	Surveillance Record for High, Locked High and Very High Radiation Area Boundary and Posting	December 28, 2001
RP-AA-210	Dosimetry Issue, Usage and Control	Revision 3
CR 00126221	RWP Exceeded Threshold Without Pre-Approved Waiver	October 10, 2002
CR 00099578	Reactor Cavity Not Posted Consistently with Drywell	March 16, 2002
CR 00126811	Worker Entered Drywell on Wrong RWP	October 10, 2002
CR 00127347	Two Individuals Locked into Posted Locked High Radiation Area	October 14, 2001

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

	Focus Area Self-Assessment Report, Radiation Protection, Dresden Station, “Outage Readiness and Preparation”	October 11 - 13, 2002

	Focus Area Self-Assessment Report, Radiation Protection, Dresden Station, “Access Control to Radiological Significant Areas” and “ALARA Planning and Controls”	September 24 - 27, 2002
RP-AA-401	Operational ALARA Planning and Controls	Revision 2
RP-AA-400	ALARA Program	Revision 2
RP-AA-4002	Radiation Protection Refuel Outage Readiness	Revision 0
	D3F17 Dose Performance Reports	October 7-11, 2002
	Plan of the Day; dated October 7-11, 2002	
	BWR Owners Group, Radiation Protection/ALARA Committee, Plant Status Report for Dual Unit Site, Dresden	July 8, 2002
	RWP (Radiation Work Permit) # 10001595; 3R17 In-Vessel Inspection Activities	Revision 0
	RWP # 10001595 (ALARA Plan); D3R17 In-Vessel Inspection Activities	Revision 2
	RWP # 10001593; D3R17 Reactor Disassembly/Reassembly and Related Activities	Revision 0
	RWP # 10001593 (ALARA Plan); D3R17 Reactor Disassembly/Reassembly and Related Activities	Revision 2
	RWP # 10001593 (Total Effective Dose Equivalent (TEDE) ALARA Evaluation Screening Worksheet); D3R17 Reactor Disassembly/Reassembly and Related Activities	Revision 1
	RWP # 10001594; D3R17 Reactor Steam Dryer Modification Diving Activities	Revision 1
	RWP # 10001594 (ALARA Plan); D3R17 Reactor Steam Dryer Modification Diving Activities	Revision 2
	RWP # 10001620; Drywell Structural Steel Modification in Support of EPU	Revision 0
	RWP # 10001620 (ALARA Plan); Drywell Structural Steel Modification in Support of EPU	Revision 2
	RWP # 10001603; D3R17 Turbine Generator - Auxillary System Maintenance	Revision 1
	RWP # 10001603 (ALARA Plan); D3R17 Turbine Generator - Auxillary System Maintenance	Revision 2

	RWP # 10001603 (Work-in-Progress Review); D3R17 Turbine Generator - Auxillary System Maintenance	Revision 2
	RWP # 10001603 (TEDE ALARA Evaluation Screening Worksheet); D3R17 Turbine Generator - Auxillary System Maintenance	Revision 1
	RWP # 10001603 (Post - Job Review); D3R17 Turbine Generator - Auxiliary System Maintenance	October 30, 2002
	RWP # 10001545; D3R17 Scaffolding Activities Transport/installation (Excluding Drywell)	Revision 1
	RWP # 10001545 (ALARA Plan); D3R17 Scaffolding Activities Transport/installation (Excluding Drywell)	Revision 2
	RWP # 10001545 (Work-in-Progress Review); D3R17 Scaffolding Activities Transport/installation (Excluding Drywell)	Revision 2
	RWP # 10001545 (Post - Job Review); D3R17 Scaffolding Activities Transport/installation (Excluding Drywell)	October 29, 2002
	RWP # 10001568; D3R17 Drywell Main Steam Isolation Valve Maintenance	Revision 0
	RWP # 10001568 (ALARA Plan); D3R17 Drywell Main Steam Isolation Valve Maintenance	Revision 2
	RWP # 10001568 (Work-in-Progress Review); D3R17 Drywell Main Steam Isolation Valve Maintenance	Revision 2
	RWP # 10001568 (Post - Job Review); D3R17 Drywell Main Steam Isolation Valve Maintenance	October 27, 2002
CR 00121620	Potential Adverse Trend in Station Exposure	September 4, 2002
CR 00125351	Outage Readiness Self-Assessment Improvements Identified	October 1, 2002
CR 00127363	Diver Receives Daily Dose Higher than Administrative Limit	October 19, 2001
CR 00127602	ALARA RWP WIP Reviews not Completed in Prescribed Time	October 20, 2002
CR 00128125	Diver Dose Attributed to Wrong RWP	October 19, 2002