April 30, 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 50-237/03-02; 50-249/03-02

Dear Mr. Skolds:

On March 31, 2003, 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 April 8, 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 eight issues of very low safety significance (Green). Six of these issues were determined to involve violations of NRC requirements. 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 power reactors 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, conduct inspections, and 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 power reactors. 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 50-237/03-002;
50-249/03-002
Attachment: NRR Response to TIA 2001-13

cc w/encl:
Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
M. Aguilar, Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission

DOCUMENT NAME: C:\ORPCheckout\FileNET\ML031200814.wpd

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RIII		RIII	E	RIII		RIII	
NAME	MRing for PPelke/trn		MRing					
DATE	4/30/03		4/30/03					

OFFICIAL RECORD COPY

<u>ADAMS Distribution</u>:
AJM
DFT
LWR
RidsNrrDipmIipb
GEG
HBC
DRC1
C. Ariano (hard copy)
DRPIII
DRSIII
PLB1
JRK1
MLD1
CGM
RBL

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-237; 50-249
License Nos:	DPR-19; DPR-25
Report No:	50-237/03-002; 50-249/03-002
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	December 29, 2002 through March 31, 2003
Inspectors:	D. Smith, Senior Resident Inspector B. Dickson, Resident Inspector R. Lerch, Project Engineer P. Pelke, Reactor Engineer J. Belanger, Senior Physical Security Inspector, DRS T. Madeda, Physical Security Inspector, DRS J. Maynen, Physical Security Inspector, DRS R. Landsman, Project Engineer, DNMS R. Schulz, Illinois Department of Nuclear Safety
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000237-03-002, IR 05000249-03-002; Exelon Generation Company; on 12/28/2002-03/31/2003, Dresden Nuclear Power Station, Units 2 and 3. Maintenance Risk Assessment, Operability Evaluations, Identification and Resolution of Problems, Event Follow-Up, and Other.

This report covers a 3-month period of baseline resident inspection, an announced baseline security inspection, and issue resolution for a decommissioning inspection. The inspection was conducted by Region III inspectors and the resident inspectors. Eight findings, six of which involved 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: Initiating Events

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, due to the licensee's failure to update a drawing for the average power range monitors. As a result, a self revealing event, a half-scram, occurred on Unit 3 during a fuse inspection activity.

  This finding was more than minor because if left uncorrected, this issue could become a more significant safety concern by resulting in an initiating event. However, because a scram did not occur, this finding was determined to be of very low safety significance. (Section 4OA3.1)

- Green. The inspectors identified one finding regarding a number of performance issues associated with the licensee's failure to properly implement vendor recommendations for the main turbine. The performance issues included improper implementation of vendor recommendations for monitoring shaft voltage, inadequate acceptance criteria for shaft voltage, and deferral of preventive maintenance.

  This finding was more than minor because it resulted in an initiating event (scram) on Unit 3. The finding was of very low safety significance because all equipment operated as designed during the scram. No violation of NRC requirements occurred as a result of the licensee's failure to adequately implement vendor recommendations for non-safety related equipment. (Section 4OA3.2)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XIV, due to licensee personnel performing work on safety related equipment without authorization and ignoring protected pathway equipment signs. This error resulted in both divisions of low pressure coolant injection/containment cooling service water (LPCI/CCSW) becoming inoperable.

This finding was more than minor because the availability of the LPCI/CCSW systems was adversely impacted and both trains were rendered inoperable as a result of this human performance deficiency. The finding was of very low safety significance because operators would easily be able to unisolate the 2B CCSW pump, all other mitigating systems were available, and the total exposure time was short. (Section 1R13.1)

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, due to the licensee's failure to promptly implement effective corrective actions upon the discovery of a generic non-conforming condition affecting a number of safety related 480 volt motor control center cubicle (MCC) auxiliary contact assemblies on both units.

  This finding was determined to be more than minor because it could be reasonably viewed as a precursor to a significant event and if left uncorrected the finding could become a more significant safety concern because the station personnel could fail to evaluate non-conforming conditions which could render safety related equipment inoperable. The finding was of very low safety significance because safety related plant equipment was not rendered inoperable as a result of the degraded condition. (Section 1R15.1)

- Green. The inspectors identified one finding regarding the licensee's preparation of an inadequate operability evaluation. The finding involved inadequacies in the licensee's documented operability evaluation for a generic non-conforming condition affecting a number of safety-related 480 volt motor control center cubicle (MCC) auxiliary contact assemblies.

  This finding was more than minor because it could be reasonably viewed as a precursor to a significant event, and if left uncorrected, the finding could become a more significant safety concern because the station could have non-conforming conditions which render safety-related equipment inoperable, even though the operability evaluations would conclude the equipment was operable. The finding was of very low safety significance because none of the safety related plant equipment was adversely affected by the non-conforming condition. Even though inadequacies were noted in the evaluation, the equipment was ultimately determined to be operable and no violations of NRC requirements were identified. (Section 1R15.2)

  Green. A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for failure to incorporate the rated live load of the RB crane into the original calculations for the reactor building (RB), a Seismic Category 1 structure. Significant NRC intervention was required over a two year period to ensure the licensee resolved the compliance and safety issues related to the qualification of the reactor building crane in a manner consistent with the Dresden licensing basis and NRC regulations.

  The finding is of more than minor significance because it affects the cornerstone attribute of design control as it relates to both the Mitigating System and Barrier Integrity cornerstone objectives. Due to the low seismic initiating event frequency, the short duration of time that the heavy loads were suspended on the RB crane, the nature of the load path and load lift controls, and the recent licensee calculations which demonstrated that the RB superstructure will support the crane in a seismic event, the findings were determined to be of very low safety significance (Section 4OA5.1).

Green. A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for the licensee's failure to ensure design stresses in roof truss members and interior building column members of the RB superstructure remained below allowable limits. Significant NRC intervention was required over a two year period to ensure the licensee resolved the compliance and safety issues related to the licensee's allowance of stress values above allowable design limits in a manner consistent with the Dresden licensing basis and NRC regulations.

The finding is of more than minor significance because it affects the cornerstone attribute of design control as it relates to both the Mitigating System and Barrier Integrity cornerstone objectives. Due to the low seismic initiating event frequency, the short duration of time that the heavy loads were suspended on the RB crane, the nature of the load path and load lift controls, and the recent licensee calculations which demonstrated that the RB superstructure will support the crane in a seismic event, the findings were determined to be of very low safety significance (Section 4OA5.2).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, due to the licensee's inadvertent entry into an unanalyzed region of the Unit 3 power-to-flow map on several occasions.

  This finding was more than minor because the licensee demonstrated inadequate reactivity management control which resulted in exceeding the Maximum Extended Load Line Limit Analysis (MELLLA) flow control line (FCL) limits on a number of occasions. This could have challenged one of the physical design barriers (fuel cladding) that protect the public from radionuclide releases. This finding was determined to be of very low risk significance because the operators did not exceed any thermal limits on the unit. (Section 4OA2)

B. Licensee Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Unit 2 began the inspection period at 912 MWe (95 percent thermal and 100 percent of rated electrical capacity). On January 10, 2003, operators reduced load on Unit 2 to 525 MWe to replace several scram solenoid pilot valves. The unit was returned to full power operations on January 13, 2003. On February 26, 2003, operators reduced power to 710 MWe to repair a leaking seal cooling line on the 2C reactor feedwater pump. The unit was returned to full power the next day.

Unit 3 began the inspection period at 912 MWe (95 percent thermal and 100 percent of rated electrical capacity). On January 13, 2003, operators reduced load to 755 MWe on Unit 3 to perform control rod drive timing and the unit was returned to full power the same day. On January 17, 2003, operators reduced load to 550 MWe to perform power suppression testing to locate the fuel leak on the unit. Also, work was performed on the 3A reactor feed pump, the 3A condensate/condensate booster pump, and the 3D condensate/condensate booster pump. The unit was returned to full power operations on January 20, 2003. On February 15, 2003, operators reduced load to 550 MWe to perform control rod drive exercising. The unit was returned to full power operations on February 18, 2002. Towards the end of the inspection period, on March 29, 2003, the operators commenced a forced outage on the unit. The operators reduced power to 200 MWe and took the turbine off line to enter the drywell to inspect for leakage from the drywell pneumatic system. By March 31, the licensee had put the turbine back on line and the unit had returned to 60 percent power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

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:

- 2A & 2B low pressure coolant injection/containment cooling service water system;
- Unit 3 125 Vdc battery system; and
- Unit 3 250 Vdc battery system.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors toured plant areas 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 3 emergency diesel generator, fire zone 9.0.B;
- Unit 2 west low pressure coolant injection system corner room, fire zone 11.2.1;
- Unit 2/3 turbine deck, fire zone 8.2.8.A;
- Unit 3 250 Vdc battery room, fire zone 6.1;
- Unit 2 250 Vdc battery room, fire zone 7.0.A.0;
- Unit 2 isolation condenser floor, fire zone 1.1.2.5.A;
- Unit 3 standby liquid control floor, fire zone 1.1.1.5.D;
- Unit 2/3 emergency diesel area, fire zone 9.0.C;
- Unit 2/3 day tank area, fire zone 3.A.2.1;
- Unit 2 emergency diesel generator room including diesel day tank room, fire zone 8.2.A.1;
- Unit 3 reactor building ground floor, fire zone 1.1.1.2;
- Unit 2 standby gas treatment system area level 538'; fire zone 8.2.6.B
- Penetrations F-46-05, F-54-05, F-56-02/03, F-72-01, F-131-06, F-134-06, F-139-19, F-142-05, and F-143-03/04; and
- Unit 2 standby gas treatment area, fire zone 3.A.1.A.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On March 12, 2003, the inspectors observed Crew# 5 in simulator training. The scenario consisted of a power reduction, localized flooding, control rod pump and condensate pumps trip, one main steam line failure to open, and emergency depressurization and reactor pressure vessel recovery.

b. No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

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:

- Standby coolant supply system;
- 250 Vdc battery system; and
- Station blackout diesels system.

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:

- Unit 2 C&D low pressure coolant injection system/containment cooling service water Division II planned maintenance;
- 3A and 3B containment cooling service water pump discharge piping replacement;
- Unit 2 high pressure coolant injection gland seal leakoff condenser maintenance;
- Unit 2 emergency diesel generator 6 year preventative maintenance;
- Unit 3 core spray system planned testing;
- Division 1 low pressure coolant injection system;
- 6 year preventive maintenance for high pressure coolant injection system; work included: installation of high pressure coolant injection area temperature switch, replacement of auxiliary oil pump control switch, and modification of MOV 2-2301-8 opening logic and performance test (SP03-01-001); and
- Unit 2/3 emergency diesel generator planned maintenance.

b. Findings

.1) Maintenance Workers Perform Unauthorized Work on 2B Containment Cooling Service Water Pump

One Green finding involving a Non-Cited Violation was identified. The finding involved the licensee's failure to ensure maintenance mechanics did not perform work on safety related equipment, which had been flagged with a protected pathway sign, without obtaining proper authorization from operations personnel.

On January 13, 2003, the licensee entered a number of technical specifications for planned maintenance work on Division II low pressure coolant injection and containment cooling service water (LPCI/CCSW) systems. The performance of this work placed Unit 2 in an elevated risk condition (Yellow). The maintenance activity consisted of replacing the discharge piping elbows on the 2C and 2D CCSW pumps and was governed by the maintenance work instructions. The licensee had placed protected pathway barriers in front of the Division I LPCI/CCSW pumps (2A & 2B) to ensure work was not initiated on the wrong equipment. However, without requesting authorization, a maintenance mechanic attempted to remove a Chicago fitting from the 2B CCSW pump, part of the Division I train of LPCI/CCSW, to facilitate draining air lines on the 2C CCSW pump. Even though the mechanic observed the protected pathway barrier, he rationalized that his actions to remove the Chicago fitting were acceptable because an isolation valve existed between the CCSW discharge piping and the Chicago fitting. However, in removing the fitting, the drain line piping containing the isolation valve and the Chicago fitting severed from the CCSW discharge piping and created a one inch hole in the discharge piping line.

As a result, operations personnel isolated the 2B CCSW pump as required by Technical Requirements Manual 2.4.a due to ASME Class III piping leakage. Once the 2B CCSW pump was isolated while both Division II CCSW pumps (C & D) were out-of-service for maintenance, the configuration resulted in the inoperability of both Divisions of LPCI/CCSW systems. The mechanic's decision to work outside of the written work instructions, by disassembling the 2B CCSW pump without obtaining proper authorization and disregarding the protected pathway sign, was considered a violation.

The inspectors used IMC 0612, Appendix B, to disposition this issue and determined that it was more than minor because the issue was associated with the Reactor Safety cross-cutting attribute of Human Performance and affected the Mitigating Systems objective to ensure the availability of the LPCI/CCSW systems. Both Divisions of LPCI/CCSW were rendered inoperable as a result of this human performance deficiency.

The inspectors evaluated this issue using Inspection Manual Chapter 0609,"Significance Determination Process." The inspectors conducted a Phase 1 screening and determined that a Phase 2 evaluation was required since the safety function of the LPCI/CCSW systems was actually lost.

The inspectors used the risk-informed inspection notebook for Dresden Nuclear Power Station, Units 2 and 3, Revision 1, dated May 3, 2002, to complete the Phase 2 evaluation. The inspectors determined that the exposure time was less than 3 days since the 2B CCSW pump was repaired and returned to service within approximately two hours after isolating the pump. For each significance determination process worksheet completed, the inspectors assumed that all mitigating capability was available except for LPCI in the suppression cooling mode. The inspectors provided credit for operator action to unisolate the 2B CCSW pump, under accident conditions, to ensure the required flowrate of 5000 gallons per minutes (gpm) would be provided by both Division I LPCI/CCSW systems. The inspectors determined that even with the 2B CCSW pump having a one inch hole, the required design basis flowrate could still be achieved by the 2A and 2B CCSW pumps, since each pump has a flow capacity of 3500 gpm.

Due to the loss of the LPCI/CCSW systems, the inspectors evaluated eleven accident sequences. Worksheet results ranged from 7 to 15 points. The most dominant core damage sequences involved: (1) Inadvertent or Stuck-Open Relief Valve with containment venting available for containment heat removal; and (2) Inadvertent or Stuck-Open Relief Valve with late inventory available for containment heat removal. The inspectors concluded that the final significance determination process result for this finding was 7 points. Therefore, this finding was considered to be of very low risk significance (Green).

Also, the inspectors reviewed the licensee's completed Apparent Cause Evaluation (ACE) #139445 for this event. The inspectors concluded that the licensee's identified corrective actions had focused on the maintenance department's lack of understanding of the restrictions imposed by the "protected pathway equipment" sign, as opposed to emphasizing that operations' authorization must be obtained prior to performing any maintenance on safety related plant equipment. Licensee management agreed with the inspectors and re-opened the ACE.

Criterion XIV, of 10 CFR 50, Appendix B, requires that measures shall be established for indicating the operating status of structures, systems, and components of the nuclear power plant such as by tagging valves to prevent inadvertent operation. Performing work on equipment flagged by a protected pathway sign and not obtaining proper authorization to work on the equipment were considered violations. However, because of its low safety significance and because it was entered into the corrective action program as CR 00139445, the NRC is treating this issue as a Non-Cited Violation (**NCV 50-237/03-002-01**), consistent with Section VI.A.1 of the NRC's Enforcement Policy.

2) <u>Sample Welding Inspection Program</u>

On 1/31/03 the inspector reviewed Work Package No. 00528439-01. The work package involved safety-related work for a piping replacement (1 inch schedule 80) on the high pressure coolant injection drain pot line. While examining the welding record the inspectors discovered that the pre fit-up, fit-up, root weld, and final weld inspections by Quality Verification Inspectors had all been waived on 1/07/03. Further investigation revealed that Condition Report No. 00133407 documented the fact that as of 12/02/02, Dresden had implemented a sampling welding inspection program in which Quality Verification Inspectors need only inspect a small sample (10%) of safety related welds for acceptance, instead of the current 100%. This sample inspection program (10% or less of all safety related welds) included a sample of pre fit-up, fit-up, root weld, and final weld inspection. The licensee stated the sample inspection program was based on ANSI/ASQC Z1.4-1993, "Sampling Procedures and Tables for Inspection by Attributes." The sampling program was described in a licensee document entitled NO-AA-300-001-1002, "Quality Verification Welding Performance Monitoring Program." and the licensee contends that the sample inspection program meets the requirements of ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities."

The sampling program was only applicable to welding activities performed by Dresden Mechanical Maintenance Department (MMD) welders as opposed to contract welders and was primarily based on training administered to the welders, peer checks by other

MMD welders, and a trial period involving a substantial number of welds completed by MMD welders where no mistakes in the welds were identified.

The inspectors developed substantial concerns that this new welding sample inspection program was not based on the risk significance of the particular welds in question and may not meet the requirements of several specifications, procedures, regulations, and standards including:

- United States of America Standard (USAS) B31.1.0-1967, which governs much of the welding at Dresden;
- ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities"
- 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Facilities"
- Exelon Procedure, NO-AA-300-001-1001, "Nuclear Oversight Independent Inspection Plan"
- ASME Boiler and Pressure Vessel Code, Section III
- Owner's Design Specification K-2202, "Specification For Piping System - Dresden Units 2 and 3"
- ANS 3.2, "Administrative Controls and QA for the Operational Phase of Nuclear Power Plants," 1988

The inspectors intend to seek assistance from NRC welding specialists and from the Office of Nuclear Reactor Regulation (NRR) to determine if the licensee's sampling inspection program is acceptable. This issue is considered an Unresolved Item (**URI 50-237; 249/03-002-02**) pending review with the above groups.

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 3 spent fuel multiple purpose canister (MPC) 68-036 and MPC 68-037 closure rings (OE #002-16);
- General Electric CR105X auxiliary contacts (OE #03-003, Revision 0);
- General Electric CR105X auxiliary contacts (OE #03-003, Revision 1 and 2);
- Safety limit minimum critical power ratio (OE #03-001, Revision 0 and 1);
- Containment atmosphere monitoring heat trace (OE #03-002, Revision 0 and 1);
- Misalignment of the Unit 2 emergency diesel generator suction piping (OE 03-004); and
- Main steam isolation valve Belleville springs (OE 002-021).

b. Findings

.1 480 Volt Motor Control Center Cubicle Auxiliary Contact Assembly Issue

Corrective Actions Upon the Discovery of a Non-conforming Condition Affecting 480 Volt Motor Control Center Cubicles Auxiliary Contact Assemblies

One Green finding involving a Non-Cited Violation was identified. The finding involved the licensee's failure to implement effective corrective actions upon the discovery of a generic non-conforming condition affecting a number of safety related 480 volt motor control center cubicle (MCC) auxiliary contact assemblies.

On February 6, 2003, the inspectors discussed the initiation of an operability evaluation for 480 volt MCC auxiliary contact assemblies with the component specialist and the electrical design engineering supervisor. The component specialist provided the following sequence of events leading to the initiation of the operability evaluation on February 5, 2003:

> The 2/3 emergency diesel generator (EDG) fuel oil transfer pump breaker was found tripped on October 29 and December 3, 2002. The breaker was located on a 480 volt motor control center (MCC). The component specialist sent the auxiliary contact assembly (CR105X) to PowerLab for analysis and was tasked to perform an apparent cause evaluation (ACE) for the two breaker failures. The component specialist received preliminary results on December 5, 2002, which indicated that the last breaker failure was due to two elements; dried white grease (Aeroshell 7) and a potential manufacturing defect in the phenolic material of the auxiliary contact assembly which was from manufacturer lot number VA737. On December 5, 2002, the component specialist generated Condition Report 134156 which documented the results from PowerLab. Subsequently, the licensee performed a walkdown of the 480 volt MCC cubicles on January 28 and 29, 2003, to identify the affected equipment containing these two elements. However, the decision to initiate an operability evaluation for this degraded condition was not made until February 5, 2003.

The inspectors were concerned that an operability evaluation had not been generated on December 5, 2002, based on the results from PowerLab. All safety related and non-safety related 480 volt MCC CR105X auxiliary contact assemblies use Aeroshell 7; yielding the two elements which contributed to the failure of the 2/3 EDG fuel oil transfer pump. Therefore, any plant equipment containing these two aspects was potentially non-conforming and thus required an operability evaluation to verify the functionality of the equipment. The inspectors also noted that site personnel had a second opportunity to have initiated an operability evaluation on January 29, 2003, upon completing the delayed walkdown, which identified several safety related components that were non-conforming because they contained these two elements. The engineering supervisor acknowledged the inspectors' position.

The inspectors were also concerned that review of this generic non-conforming condition by the shift manager on December 5, 2002, and the management review committee members on December 9, 2002, failed to identify the generic implications of these degraded conditions on safety related equipment. The delay in conducting walkdowns to identify affected equipment for 7 weeks, from December 5, 2002 to January 29, 2003, further aggravated the lack of timely action. Therefore, the inspectors concluded that the licensee's actions were not commensurate with safety. The results of the walkdown inspections are discussed in Section 1R15.2 of this inspection report.

The inspectors used Inspection Manual Chapter 0612 to disposition this issue of the licensee's failure to promptly identify and correct conditions adverse to quality when station personnel were made aware of a generic non-conforming condition potentially affecting safety related plant equipment. The inspectors determined that this issue was more than minor because the finding could be reasonably viewed as a precursor to a significant event and if left uncorrected the finding would become a more significant safety concern because the station personnel could fail to evaluate non-conforming conditions which could render safety related equipment inoperable. The finding was associated with the reactor safety cross-cutting attribute of human performance and impacted the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (Green).

10 CFR 50, Appendix B, Criterion XVI, requires that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this, the licensee failed to promptly implement effective corrective actions associated with the discovery of a generic non-conforming condition, consisting of dried white lubricant along with a potential manufacturing defect, which affected multiple safety related 480 volt motor control center auxiliary contact assemblies. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (**NCV 50-237; 249/03-002-03**) consistent with Section VI.A.1 of the NRC Enforcement Policy. This issue was entered into the licensee corrective action program as CR145061.

.2 Review of Operability Evaluation for 480 VAC Breakers Containing CR105X Auxiliary Contacts

One Green finding for the preparation of an inadequate operability evaluation was identified. The finding involved the licensee's documented operability evaluation for a generic non-conforming condition affecting a number of safety-related 480 volt motor control center cubicle (MCC) auxiliary contact assemblies.

As stated in Section 1R15.1 of this report, the licensee identified that there was a generic plant issue associated with a possible molding defect with the CR105X auxiliary contacts used in safety-related breakers at Dresden and the lubricant used on the auxiliary contact assemblies. The inspectors reviewed OE 03-003, Revisions 0 and 1 and noted several deficiencies. The following inadequacies were noted by the inspectors.

A) Inappropriately Narrowed Focus on Date Code of Auxiliary Contact Assembly

During initial troubleshooting for the second breaker failure of the 2/3 emergency diesel generator fuel oil transfer pump, the licensee noted the stamped date code (VA737) on the failed auxiliary contact assembly. A test lab's analysis report (DRE 43266) attached to OE 03-003 concluded the following, "Wearing of the phenolic plunger was possible due to an original molding defect and build up of the Aeroshell 7 grease thickener residue contributed in the binding of one of the ganged auxiliary contact assemblies. Upon removal of the phenolic dust residue and magnesium silicate residue along the plunger rail, the auxiliary contact operated properly." In Section 2.3 of the operability evaluation, the licensee identified that ten safety-related breaker cubicles contained CR105X auxiliary contact assemblies with dried grease. However, the assemblies did not contain the VA737 date code. These breakers were listed in Attachment 2 of OE 003-03. The OE specified that since these safety-related breakers did not have the affected date code, the operation of the auxiliary contact assemblies was expected to be good and they would be replaced during the next maintenance window. The inspectors concluded that this statement was unfounded since the licensee was unable to provide operating history or vendor information to conclude that only the auxiliary contact assemblies date stamped VA737 were affected by the generic molding defect. The licensee acknowledged the inspectors' conclusions, and the affected auxiliary contacts were replaced within 2 weeks. The licensee revised the operability evaluation to reflect the inspectors' position.

B) Lack of Rigor in Determining Extent of Condition of Use of CR105X Auxiliary Contact Assemblies

The operability evaluation only addressed 480 volt safety-related breakers. The inspectors concluded that the licensee had not fully determined the extent of condition because none of the existing deficiency information limited the use of the CR105X auxiliary contact assemblies to 480 volt breakers only. However, revisions 0 and 1 of the operability evaluation did not address whether or not the 220 volt breakers were affected by this issue. The licensee acknowledged the inspectors' concerns. The licensee performed an inspection of the 220 volt breakers and discovered CR105X auxiliary contact assemblies were used in several of the breakers. The licensee identified that none of these CR105X auxiliary contact assemblies contained dried grease. The licensee subsequently revised the operability evaluation to reflect this information.

C) Lack of Documented Operability Determination for a Primary Containment Isolation Valve

In Revision 0 of OE 003-03 the licensee stated that successful performance of the affected equipment during surveillance tests or continued operation of these components constituted reasonable assurance of the operability of the safety-related equipment associated with the subject MCCs. During the walkdown of the 480 volt breakers, the licensee identified that the breaker for the reactor building closed cooling water (RBCCW) drywell return header inboard isolation valve (motor operated) contained a CR105X contact with dried grease. The valve was a primary containment isolation valve which was normally open and only cycled closed at the onset of refueling outages. The safety function of this valve was to close for drywell isolation. The inspectors noted that the last refueling outage for Unit 2 was more than 450 days prior to completing the operability evaluation. On February 18, 2003, the inspectors expressed concern to the licensee that this primary containment isolation valve did not meet the criteria for operability set forth in OE 003-03, yet had no other documented justification for operability. The licensee understood the inspectors' concern. The auxiliary contact was manually exercised the following day, with acceptable results, to confirm operability. The licensee then replaced the auxiliary contact.

The inspectors concluded, based on the number of deficiencies identified during their independent review of the operability evaluation, that the evaluation lacked information to prove component functionality or operability. The inspectors used Appendix B, Inspection Manual Chapter 0612 to disposition this issue. Based on the potential implication of multiple degraded safety-related systems, the inspectors determined this issue was more than minor. The finding could be reasonably viewed as a precursor to a significant event, and if left uncorrected, the finding could become a more significant safety concern because the station could have non-conforming conditions that may render safety-related equipment inoperable. The finding was associated with the reactor safety cross-cutting attribute of human performance and impacted the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (Green). Even though inadequacies were identified in the operability evaluation, all of the affected equipment was ultimately determined to be operable and no violations of NRC requirements were identified. (**FIN 50-237;249/03-002-04**)

1R16 Operator Work-Around (71111.16)

a. Inspection Scope

The inspectors reviewed operator work-around #03-OB-42, "Service water system pressure decrease," 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.

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:

- Replaced scram solenoid pilot valve for control rod drive F-2;
- Added packing to air operated valve (AOV) 2/3-5741-48B, service water supply to control room air conditioning unit B;

- Replaced 480 volt breaker auxiliary contacts for motor operated valve (MOV) 2-1301-4 and MOV 2-1501-32B;
- Replaced Unit 3 high pressure coolant injection auxiliary oil pump motor;
- Replaced Unit 2/3 emergency diesel generator cooling water pump;
- Replaced Unit 2 high pressure coolant injection system cooler room; and
- Replaced Unit 2/3 diesel oil transfer pump.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

On March 29, 2003, the licensee began a forced outage on Unit 3. The operators took the turbine offline and inserted all control rods to enter the drywell to search for leakage in the drywell pneumatic system.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed surveillance testing on risk-significant equipment and reviewed test results. The inspectors assessed whether the selected plant equipment could perform its intended safety function and satisfy the requirements contained in technical specifications. Following the completion of the test, the inspectors determined that the test equipment was removed and the equipment returned to a condition in which it could perform its intended safety function.

The inspectors observed surveillance testing activities and/or reviewed completed packages for the tests, listed below, related to systems in the Initiating Event, Mitigating Systems and Barrier Integrity Cornerstones:

- Unit 2 reactor vessel level indication system;
- Unit 2/3 core height level transmitter calibration;
- Unit 2 electro hydraulic control system;
- Unit 2 turbine control system;
- Unit 3 containment cooling service water system;
- Unit 3 low pressure coolant injection system; and
- Unit 3 core spray system.

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)# 340485, “Unit 2 Temporary Modification to Jumper 250 VDC Battery Cell #10"; and
- Engineering evaluation (EC)# 336396, “Gag Open Reactor Feed Pump Ventilation Temperature Control Damper.”

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP4 Security Plan Changes (71130.04)

a. Inspection Scope

The inspectors reviewed Revision 68 to the Dresden Nuclear Power Station Security Plan and Revision 68 to the Dresden Safeguards Contingency Plan to verify that the changes did not decrease the effectiveness of the submitted documents. The referenced revisions were submitted in accordance with regulatory requirements by licensee letter dated February 4, 2003.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Initiating Events

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 Unplanned Scrams (Fourth Quarter 2001 through Fourth Quarter 2002); and

- Unit 2 and Unit 3 Scrams with Loss of Normal Heat Removal (Fourth Quarter 2001 through Fourth Quarter 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 three 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:

- Maximum extended load line limit analysis (MELLLA) boundary;
- Standby coolant supply valve; and
- Untimely and incorrect coding of condition reports.

b. Findings

.1 Multiple Occurrences of Exceeding the Maximum Extended Load Line Limit Analysis (MELLLA) Limit

One Green finding and an associated violation were identified following an inadvertent entry into an unanalyzed region of the Unit 3 power-to-flow map during a down power using reactor recirculation flow on January 8, 2003.

Section 5.6.5 of Dresden's Improved Technical Specifications required that core operating limits be established prior to each reload cycle, or prior to any remaining portion of a reload cycle and that they be documented in the Core Operating Limits Report (COLR). The COLR and any subsequent revisions are submitted to the NRC for review. The licensee used an analysis that looked at each design basis accident from bounding initial conditions to establish these operating limits. The bounding initial conditions are at the maximum rod line with varying reactor core flow as depicted on the licensee's power-to-flow map. The MELLLA was used to develop the upper limits of the power-to-flow map. The upper limit of the power-to-flow map was called the MELLLA flow control line (FCL) limit. At Dresden, the MELLLA FCL limit was 103.2 percent.

On Jan 8, 2003, during a planned load drop from 912 MWe to 800 MWe, the Dresden Unit 3 FCL increased to approximately 105 percent. The nuclear engineering personnel had predicted the FCL value would increase from 100.1 to 101.0 percent. After recognizing the plant was above the MELLLA FCL limit of 103.2 percent, operators appropriately decreased the FCL to 101 percent by increasing the electrical output of the unit back to 912 MWe and verified that all fuel thermal limits remained bounded by analysis.

Following this event the staff performed a prompt investigation and a root cause investigation. These investigations identified a number of programmatic and human performance deficiencies, including the following:

- Inadequate implementation and communication of changes in operational strategy as a result of individual accountability and behavior issues associated with the execution of corporate reload control procedures;

- Insufficient guidance for predicting core response to reactivity changes;

- Use of inadequate indications for monitoring reactor power and flow during operation near or above the MELLLA boundary;

- Failure to use all available tools such as OD3 (heat balance) output deck results and the electronic power-to-flow map on the computer, as power was maneuvered; and

- Inadequate pre-job briefing which failed to discuss the use of the power-to-flow map, proper communication between the qualified nuclear engineer and operations, and roles and responsibilities prior to the reactivity change.

The root cause report also stated that there were several instances leading up to and during the event where operation, reactor engineering, and nuclear fuel personnel did not exhibit the appropriate amount of sensitivity for reactivity management.

The instances included:

- On May 2, 2002, the MELLLA FCL Limit was exceeded during control rod manipulation on unit 2 due to reactor engineering personnel inadequately accounting for Xenon conditions. The licensee's corrective action included a review of the event in licensed operator requalification training and revision to the procedure governing routine power changes.

- On July 6 and 11, 2002, the MELLLA FCL limits were exceeded (103.3 - vs - 103.2), during FCL rod manipulation. The licensee did not implement any corrective actions for this event.

- On August 7, 2002, the MELLLA FCL limit was momentarily exceeded during power reduction using reactor recirculation flow. Again, the licensee did not implement any corrective actions in response to this event.

- November 1, 2002, during withdrawal of a control rod, the FCL was thought to have exceeded the MELLLA FCL limit because the control room alarm annunciated. Follow-up calculation verified that the MELLLA boundary had not been exceeded. The cause of the event was an inadequate review of the input screen to predict core response and a larger than expected increase in FCL for the rod pulls. The licensee implemented corrective actions included reinforcing expectations for using human performance prevention tools for the reactor engineering personnel involved with the event. Additionally, reactor engineering personnel were assigned an action item to create a power-to-flow map which would allow operators to visually track the FCL during reactivity manipulations.

The inspectors determined that the issue of exceeding MELLLA limits was related to a licensee performance deficiency. This issue resulted in operating in an unanalyzed region of the power-to-flow map which was foreseeable and preventable by the licensee.

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 determined that the finding was associated with the configuration control, procedure quality, and human performance attributes of the barrier integrity cornerstone. The inspectors also determined that the finding affected the cornerstone objective of providing reasonable assurance that physical design barriers (e.g. fuel cladding) protect the public from radionuclide releases caused by accidents or events since operation outside established operational limits, such as the power-to-flow map, can lead to a violation of thermal limits and fuel cladding damage. As a result, the inspectors concluded that the finding was more than minor. This issue screened out as Green using Phase 1 of the significance determination process because no thermal limits for the unit were actually exceeded.

Criterion XVI, "Corrective Action," of 10 CFR 50, Appendix B, requires that conditions adverse to quality, such as deficiencies and deviations, be promptly identified and corrected. Based on the repeat instances of exceeding the MELLLA FCL limits, the inspectors concluded that corrective actions taken in response to these events were inadequate. The inspectors also concluded that exceeding the MELLLA FCL limit was an event which affected quality, since the MELLLA FCL limit was used to establish core thermal limits in the COLR. Therefore, the failure to prevent operation outside an analyzed region of the power-to-flow map on January 8, 2003, was an example where the requirements of 10 CFR 50, Appendix B, Criterion XVI, were not met and was a violation. However, because of its low safety significance and because it was entered into the corrective action program, the NRC is treating this issue as a Non-Cited Violation (**NCV 50-249/03-002-05**), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. The issue was entered into the licensee's corrective action program as CR 138515.

.2 Maintenance Rule Functional Failures Not Previously Identified in Standby Coolant Supply Valve

On February 6, 2003, the inspectors had discussions with the system engineer regarding condition reports (CRs) 142284 and 142281. The CRs documented that maintenance problems with the unit 2 and unit 3 standby coolant supply valves were conservatively declared as maintenance rule functional failures (MRFFs). The system engineer informed the inspectors that the CRs were the result of an apparent cause evaluation (ACE) that was completed in October 2002. The purpose of these valves is to provide water to the hotwell for condensate and feedwater for core and containment flooding.

The system engineer was assigned ACE 116219 to determine the reliability of the standby cooling supply valves and to determine if any MRFF had occurred, based on comments by operations personnel regarding the unreliability of these valves during outage testing. The ACE concluded that MRFFs were believed to have occurred and that maintenance personnel did not fully understand the criteria for generating CRs as specified in the maintenance department's administrative procedure for writing CRs. The ACE documented that work order (WO) 373122 was written to address the thermal overloads tripping when cycling the valve in both directions. The tripping of the breaker occurred after a preventive maintenance task, lubricating the stem, was performed on the valve. Also, the ACE specified that WO 373122 had been previously reviewed by the system engineer in May 2002, as part of the station's review effort for 1700 work requests which had not been reviewed for MRFF determinations. The ACE did not address why WO 373122 was not identified as a MRFF by the system engineer during that review.

In addition, a quality review team (QRT), responsible for reviewing engineering products, reviewed ACE 116219 and determined that the ACE had several deficiencies. The QRT noted several minor administrative deficiencies and that the ACE did not fully address the issue of determining if historical MRFFs of the valves occurred. The QRT noted that the ACE indicated preconditioning of the valves during the surveillance testing because the preventative maintenance activity was performed prior to surveilling the valve. However, the ACE did not address this deficient aspect. Also, the QRT did not identify that since this failure was a MRFF per the maintenance rule administrative procedure, the extent of condition of the May 2002 work request review needed evaluation. To address the concerns raised by the QRT, the system engineer provided a summary of changes to the ACE that addressed MRFF and procedure inadequacies and noted that two MRFFs had occurred.

The inspectors had the following concerns with this issue: 1) the system engineer failed to identify an October 2001 failure as a MRFF during the May 2002 review; 2) the MR coordinator failed to identify this oversight by the system engineer; 3) the QRT failed to recognize the quality of the May 2002 work request review was suspect due to this new MRFF determination; and 4) the QRT did not challenge the system engineer's response which did not address the aspect of preconditioning the valve.

On February 21, 2003, the licensee informed the inspectors that system engineers would re-review all 1700 service work request because the station did not have assurance that the 2002 work requests review had been properly conducted since all the system engineers used the same criteria to determine MRFFs.

.3 Inadequate Knowledge of Emergency Diesel Generator Governor Oil Level

On February 27, 2003, the inspectors identified that the oil level in the unit 2/3 emergency diesel generator was either at or below the designated oil band lower level markings on the governor sight glass. The inspectors informed on-shift operators. The operators informed the inspectors that according to operating procedures, the diesel governor oil level was adequate as long as there was oil in the sight glass. The system engineer concurred with the on-shift crew. The inspectors disagreed with this position. The operators also stated that a post maintenance test would be required if oil was added to the governor. The inspectors reviewed the vendor's manual (Manual 03040D) for the UG-8 Woodward Dial Governor. In Manual 03040D, the vendor stated the correct oil level was 0.75 to 1.25 inches below the top of the governor case. These levels corresponded to the upper and lower level markings of the oil band on the sight glass, as verified by both the inspectors and the system engineer. Additionally, the system engineer confirmed that adding oil to the governor did not require post maintenance testing.

This issue had been previously identified by the inspectors in January 2002. During this time, the inspectors identified that the Unit 3 emergency diesel generator was either at or below the designated oil band lower level markings on the governor sight glass. The licensee operations staff provided a response that was similar to the recent response. In 2002, the inspectors presented the same vendor information to the licensee which specified an oil level band. The licensee acknowledged inspector concerns, however, did not change or revise the appropriate procedures. On both occasions the system engineer stated the governor oil level was at the lower oil band. The system engineer explained that the inspectors' identification that the oil level was below the lower oil level band was caused by parallax (alignment distortion). The inspectors were standing at a slight angle to the sight glass and this distortion caused the oil level to appear below the lower level band marking. The inspectors and system engineer reverified that the oil level was at the lower oil level marking. The licensee also immediately took action to replenish oil level in Units 2, 2/3 and 3 emergency diesel generators. The inspectors concluded that the failure to adequately address this issue through the corrective action process in 2002 could have resulted in the emergency diesel generator being left in an undesirable condition.

.4 <u>Untimely and Incorrect Coding of Condition Reports (CRs)</u>

Several CRs were generated in a very untimely manner during the quarter even after prompting by the residents that plant deficiencies existed. Also, the CRs were incorrectly coded as licensee identified deficiencies. The issues included:

A) <u>During Turbine Control Valve (TCV) Testing the Licensee Inserted the Wrong Penalty When the #4 TCV Failed</u>

On March 4, 2003, during turbine control valve surveillance testing, the Unit 2 main turbine number 4 control valve failed its surveillance test criteria. The licensee's immediate investigation determined that the cause of the failure was due to the valve's fast-acting solenoid not responding to the test actuation signal. The design function of the fast-acting solenoid was to immediately dump the electro hydraulic control oil (EHC) from the turbine control valve under a main turbine generator load reject signal (power/load unbalance). A pressure switch located in the valve casing will sense the loss of EHC oil pressure and send a scram input signal to the reactor protection system (RPS). The scram input signal to the RPS system was credited in the licensee's core operating limits report (COLR) as an anticipatory scram to preclude the high pressure/high neutron flux scram that would be generated from the sudden closure of the main turbine control valves.

As a result of this failure, the licensee decided to take a thermal limit penalty for turbine control valve slow closure to ensure thermal limits remained bounded under accident conditions. This penalty was credited in the COLR for the failure of a turbine control valve to fast close for all reasons except under a load unbalance event. The inspectors questioned the licensee on whether the correct penalty had been inserted on the unit. On March 7, 2003, the licensee decided to switch to the thermal limit penalty for power/load unbalance signal being out of service. The licensee failed to generate a condition report documenting the reason for this change until prompted by the inspectors. The inspectors also noted that the condition report documenting this issue did not identify that this penalty change was partly due to NRC questioning.

B) <u>Technical Specification 3.0.3 Entry for Two Potentially Inoperable Core Spray Pumps</u>

On February 19, 2002, during the performance of the 24-month surveillance test for Unit 3 core spray, the licensee failed to properly restore the 3A core spray pump to service following testing. The onshift crew failed to perform a bump check of the 3A core spray pump, as specified by procedure, before preceding to test the 3B core spray pump. Once this error was identified, the onshift operators generated a CR. However, as a result of the plant conditions consisting of both core spray divisions potentially having been inoperable for eleven hours, the inspectors believed the licensee should have entered the Technical Specification (TS) statement 3.0.3. The 3B core spray pump was rendered inoperable for testing at approximately 5:00 p.m. on February 19, 2003, but the 3A core spray pump was not subsequently bump checked until 4:12 a.m. on February 20, 2003.

The inspectors discussed need to have entered Technical Specification (TS) 3.0.3 for potential inoperability of both core spray pumps with operations management. Operations management disagreed with the inspectors. After several discussions, operations personnel decided to revise the control room log on February 22 to elaborate on the shift manager's thought process for not entering TS 3.0.3. A condition report was not written to address the onshift crew's decision to not enter TS 3.0.3 until 13 days later.

C) <u>Untimely Documentation</u>

480Volt Motor Control Center Auxiliary Contact Assemblies

This issue was previously discussed in Section 1R15.1 of this inspection report. On February 6, 2003, the inspectors discussed the station's untimeliness in responding to the generic non-conforming condition with the 480 volt motor control center (MCC) auxiliary contact, which was identified in December 2002, with site management. A condition report documenting the untimely response by the site was not written and processed through the management review committee until February 27, 2002. The licensee was on day two of the station three day time limit for preparing an operability evaluation for this non-conforming condition.

Missed Maintenance Rule Functional Failures

This issue was previously discussed in Section 4OA2 of this inspection report. On February 13, 2002, the residents had discussions with plant management regarding the station's failure to evaluate the extent of condition of the May 2002 work request review after identifying MRFF determinations had been missed. The licensee informed the inspectors that the review would be completed by April 25, 2003, and that a CR would be written for the failure to properly evaluate the extent of condition for the May 2002 review and for the potential issue of pre-conditioning of the valves. The licensee did not generate CR 146675 until February 27, 2003.

Multiple Occurrences of Exceeding the MELLLA Limit

This issue was previously discussed in Section 4OA2.1 of this inspection report. On various occasions, onshift crews exceeded the MELLLA limit during reactivity management manipulations without corrective actions being implemented because either CRs were not written for the events or ineffective corrective actions were taken for the events.

4OA3 Event Follow-up (71153)

.1 Unexpected Half Scram On Unit 3 During Fuse Inspection

a. Inspection Scope

The inspector reviewed the circumstances surrounding a Unit 3 unexpected half scram which occurred on January 23, 2003, following the commencement of licensee fuse inspection activities.

b. Findings

A finding of very low safety significance (Green) and an associated Non-Cited Violation were identified for the licensee's failure to update the average power range monitor system drawing. This finding was a self revealing event due to the automatic occurrence of an unexpected half scram.

On January 23, 2003, maintenance personnel were performing operating power range monitor fuse socket verification when an unexpected half-scram was received in the Unit 3 control room. The licensee's prompt investigation of this event found the power supply to the average power range monitor flow convertor was inadvertently removed from service causing the unexpected half-scram.

The licensee performed an apparent cause evaluation (ACE) which found that the drawings used by the maintenance personnel did not depict a modification to the power supply circuitry for the average power range monitor flow convertor. The modification was for installing the phase II operating power range monitor and was completed during the Unit 3 October 2002 refueling outage.

Discussions with the engineering supervisor indicated that maintenance planners should have understood the methodology for maneuvering through the station's Passport system (licensee computer system) to obtain information which could impact system drawings. The ACE completed by the licensee discussed these methodologies; however, the ACE indicated that these methodologies were unknown to the craft, planners, and managers in the maintenance organization, were not part of the licensee's procedures, and were not part of the licensee's training program. The ACE also documented that even if the alternative methods discussed in the ACE were used in this case, there were no changes pending against the drawing.

The licensee had previously identified this problem when using these methodologies to maneuver through Passport during this same modification on Unit 2 according to the ACE. However, this issue was not documented in a condition report nor were corrective actions undertaken to ensure this condition adverse to quality was prevented from occurring again on Unit 3.

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 because if left uncorrected, this issue could become a more significant safety concern. The inspectors determined that this finding was associated with reactor safety and impacted the initiating events cornerstone by increasing the likelihood of an initiating event.

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." Using the SDP Phase 1 Screening Worksheet, this issue screened out as Green.

Appendix B, Criterion V, of 10 CFR Part 50, requires that activities affecting quality shall be accomplished in accordance with instructions, procedures, or drawings appropriate to the circumstances. On January 23, 2003, Dresden maintenance personnel were performing an inspection of the operating power range monitoring fuses and caused a half-scram condition, due to using an inadequate drawing. This 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 an Non-Cited Violation (**NCV 50-249/03-002-06**), consistent with Section VI.A.1 of the NRC's Enforcement Policy.

.2 (Closed) LER 50-249/2002-002: "Reactor Scram due to Main Shaft Oil Pump Failure"

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 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

One Green finding was identified regarding a number of performance issues associated with the licensee's failure to properly implement vendor recommendation for the main turbine which resulted in the occurrence of the scram. The performance issues included improper implementation of vendor recommendations for monitoring shaft voltage, inadequate acceptance criteria for shaft voltage, and deferral of preventive maintenance.

On July 21, 2002, Dresden Unit 3 scrammed from 100 percent power as a result of the main turbine tripping. The turbine trip was due to low main shaft oil pump (MSOP) discharge pressure. Inspection of the turbine front standard following the scram identified extensive damage to the main shaft rotor and MSOP gears. The permanent magnet generator stub shaft was found sheared. The radial and thrust bearing babbit on the generator end of the main shaft rotor was worn. The inboard MSOP bearing was worn and the anti-rotation pin was sheared. The babbit on the outboard MSOP bearing was worn. Evidence of electrolysis was visually observed and confirmed by failure analysis. Electrolysis was evident on the outboard MSOP bearing, inboard main shaft thrust bearing, and the main shaft gear teeth. The licensee concluded that the cause of the event was degradation of the auxiliary control rotor gear coupling insulation, resulting in a current flow through the shaft which caused electrolysis and accelerated wear of the bearings.

Six days prior to the scram, on July 15, 2002, the Unit 3 permanent magnet generator was observed to be discolored, and at times the edge of the housing was red hot. Metal shavings and chips were found at the outboard end with the shaft nut and washer lying on the floor. In addition, sparks were seen at the outboard end. The surface temperature was later measured at approximately 800 degrees F. The Unit 3 turbine was taken off-line. General Electric and corporate turbine personnel reviewed clearance measurements and concluded that the shaft was centered, which indicated no bearing wear at the MSOP end of the shaft. Based on the measurements taken, the front standard was not disassembled for inspection. The turbine was placed back on-line with the permanent magnet generator internals removed. Inspection upon disassembly of the turbine front standard after the July 21, 2002, reactor scram revealed bearing wear at the ten and seven o'clock positions, which would not have been revealed when the turbine was at rest.

The licensee failed to properly implement recommendations from the vendor for the main turbine. In 1984, General Electric issued TIL-973-3, "Front Standard Gears and Bearings - For Nuclear Plants," to warn of turbine front standard gear and bearing failures caused by abnormal conditions, such as shaft current flow, misalignment, and inadequate lubrication of gears and bearings. TIL 973-3 also discussed eight reported failures of gear shaft bearings accompanied by damage to journal surfaces, and three cases in which the permanent magnet generator was damaged as a result of pump gear shaft bearing failure. TIL 973-3 identified that the major cause of main shaft rotor gear and bearing failures was believed to be flow of electric current from the turbine shaft to ground. TIL 973-3 recommended that shaft voltage should be checked at least once a month with an oscilloscope and at least once a week with the remote testing device. However, Dresden procedure DES 5600-05, Revision 2, "Turbine Shaft Voltage Surveillance," implemented the monthly oscilloscope reading with incorrect acceptance criteria, and the remote testing device (installed millivoltmeter) on a monthly rather than weekly basis with no acceptance criteria.

In September 2000, as part of the D3R16 refueling outage scope controls, station system engineering and corporate engineering determined that it was acceptable to defer the 6-year Unit 3 front standard overhaul and inspection preventive maintenance until D3R17 (October 2002). This deferral was approximately 3.5 years past the due date (the preventive maintenance was last performed in May 1993).

Review of the shaft voltage readings taken per DES 5600-05 revealed an increased trend of shaft voltage levels. The voltages had increased from 25 VDC in April 2002 to as high as 100 VDC in June 2002. The recommended General Electric acceptance criteria was 1 VDC as measured from the remote shaft voltage meter. There was no acceptance criteria for this reading in DES 5600-05. The same criteria, as read on the oscilloscope, was that the maximum instantaneous voltage must not exceed 6V ground-to peak while the oscilloscope is in the DC coupled mode. DES 5600-05 acceptance criteria for this voltage measurement was 12V peak-to-peak, which did not take into account the DC offset.

Using IMC 0612, Appendix B, "Issue Dispositioning Screening, " the inspectors determined that this finding was more than minor because it affected the initiating events cornerstone objective to limit the likelihood of an initiating event. Using IMC 0609, Appendix A, "User Guidance for Significance Determination of Reactor Inspection Findings for At-Power Situations," the issue was determined to be of very low safety significance (Green) because all equipment responded as designed during the scram. No violation of NRC requirements occurred as a result of the licensee's failure to adequately implement vendor recommendations for non-safety related equipment. (**FIN 50-249/03-002-07**)

4OA5 Other

Unit 2/3 Crane Issues (60855)

Introduction

In order to resolve the licensing basis of the Reactor Building (RB) Superstructure and the crane bridge and trolley, the inspectors requested technical assistance from the Office of Nuclear Reactor Regulation (NRR). Task Interface Agreement (TIA) 2001-13 dated September 28, 2001, was issued to request review and comment on a backfit analysis related to the long term use of the Unit 2/3 RB crane to lift heavy loads at the Dresden Nuclear Power Station. In order to respond to questions related to the original licensing basis of the crane, the licensee issued April 12, 2002, and July 8, 2002, responses to an NRR request for additional information.

The licensee's response referenced a new revision, Revision 1, to calculation DRE 98-0020, which analyzed the RB superstructure with the crane loaded for forces imposed from both the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE).

On February 2, 2003, NRR issued a response to TIA 2001-13. (The TIA is enclosed as an attachment to this report.) In the TIA, NRR concluded, based on the new information and calculations provided by the licensee, that compliance with the licensing basis for the RB crane established in 1976 will provide an acceptable level of safety. The NRC determined that no further backfit analysis was necessary.

This section will address previously identified compliance issues from inspection report 07200037/2001-002.

The inspectors initially identified these issues during an inspection of the Dresden dry cask storage and handling facilities. However, the safety impact of the findings regarding the Unit 2/3 crane system were primarily related to the operation of Dresden Units 2 and 3. Therefore, the Unresolved Items initially opened related to 10 CFR Part 72 will be closed using 10 CFR Part 50 criteria.

The inspectors determined that these findings were greater than minor in accordance with Inspection Manual Chapter [IMC] 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," dated February 2, 2003. The findings dealt with Mitigating Systems and Barrier Integrity Cornerstone objectives related to the attributes of design control and equipment performance. Assumptions regarding the risk significance of Dresden RB crane issues were coordinated with a Region III Senior Reactor Analyst. Bounding assumptions included a potential rapid lowering of the heavy load while over the Unit 2 suppression pool due to heavy load handling system failures. Due to the low seismic initiating event frequency, the short duration of time that the heavy loads were suspended on the RB crane, the nature of the load path and load lift controls, and the recent licensee calculations which demonstrated that the RB superstructure will support the crane with lifted load in a seismic event, the findings were determined to be of very low safety significance (Green).

a. Inspection Scope

The inspectors reviewed unresolved items from Inspection Report 07200037/2001-002.

b. Findings

.1 Closed - Unresolved Item (URI 07200037/2001-002-05): Long-term acceptability of the Unit 2/3 RB structure, the RB crane, and ancillary equipment for handling large numbers of dry fuel storage casks. This item was unresolved pending acceptability of the RB crane system under seismic conditions. The NRC reviewed the licensing basis for the crane and RB superstructure and determined that additional information and calculations were required to justify the acceptability of the crane system.

Description - RB Superstructure:

Dresden Unit 2/3 Updated Final Safety Analysis Report (UFSAR), Section 3.2.1 classified the RB as a Seismic Category 1 structure. Updated Final Safety Analysis Report Section 3.8.4.1.3 defined the load combinations for Class 1 structures with OBE and SSE, respectively as follows:

* D (dead load + live load) + E (OBE load)
* D + E' (SSE load)

Calculations of record to address OBE and SSE, from initial operation of Units 2 and 3, including DRE 98-0020, Revision 0, dated March 16, 1998, did not consider lifted loads on the crane in conjunction with a seismic event. Therefore, those calculations did not demonstrate that the building superstructure was capable of safely supporting a lifted load on the crane during a seismic event.

Consequently, the staff issued a Request for Additional Information (RAI) to the licensee dated February 26, 2002. In the RAI, the staff requested that the licensee perform an analysis of record that supported the seismic qualification of the RB crane bridge, trolley and other load bearing components. Justification was requested to demonstrate how the crane and supporting structure met the intent of Branch Technical Position (BTP) APCSB 9-1, "Overhead Handling Systems for Nuclear Power Plants," dated January 10, 1975, (found in the Standard Review Plan [SRP], NUREG-0800) including how the analysis addressed seismic loading plus maximum critical load (lifted load). Specifically, the NRC requested that the licensee provide assurance that the combined effect of the maximum lifted load with a postulated OBE event met the design criteria, i.e., the allowable stresses specified in the UFSAR.

The licensee completed additional calculations, including DRE 98-0020, Revision 1, dated April 23, 2002, to evaluate the RB superstructure for several load combinations, including pendulum effects. The combined effect of the maximum lifted load with a postulated OBE event was evaluated. Considering the maximum lifted load (crane capacity) of 125 tons in combination with the applicable loads for the OBE loading condition, the Interaction Coefficients (IC) (where IC = Applied Stress/Allowable Stress) for the RB superstructure members, connections, and anchorages as well as the runway girders were all determined to be less than or equal to 1.0 except in two areas. The superstructure interior column IC was determined to be 1.05, and the interior column base plate IC was 1.03.

Additional refined analyses were performed to determine the conditions under which the stresses in the two overstressed elements could be shown to be within allowable limits and their ICs would be less than or equal to 1.0.

Several options were explored for this purpose, namely, evaluation of actual loads of the items to be lifted instead of the maximum lifted load of 125 tons used in the initial evaluation and/or specifying a travel path for the crane which would limit the crane reach to prescribed limits such that the stresses in the affected members would be reduced to within allowable limits.

The allowable lifted load calculation, in order to remain within stress limits and with no limits on crane movements and allowable reach, showed that the allowable lifted load for the crane should be limited to 93.75 tons, which includes the weight of the lifting apparatus.

The calculation results for the allowable reach of the crane hook, with a 125 ton lifted load, showed that the crane hook maximum reach to either end of the bridge beams must be limited to a minimum of 25 feet from the runway girder. This is equivalent to 27'-3" from column lines H and N.

The calculation results for a maximum cask weight of 105 tons showed that, for the prescribed travel path and with limits on the hook allowable reach so that the minimum distance of the lifted load from end of crane girder is 23.54 ft, all the members of the RB superstructure are within allowable stress limits and the RB superstructure is structurally adequate to support the cask weight during a postulated OBE event. All ICs were found to be less than 1.0 for a cask weight of 105 tons including the lifting apparatus.

The calculation for lifting the Reactor Vessel Head, weight 125 tons, during a postulated OBE event showed that, for the prescribed travel path and with limits on the crane hook allowable reach so that the minimum distance of lifted load to the end of the crane girder is 32.79 ft, the RB superstructure is structurally adequate to support the Reactor Vessel Head weight, and ICs for all the members are less than 1.0.

The calculation results showed that the RB superstructure members were all adequate to support the shield plugs weight during a postulated OBE event. All ICs were less than 1.0 provided that the hook maximum reach while lifting the bottom layer is limited to a minimum of 11'-0" from the runway girder. This is equivalent to 13'-3" from column lines N and H. Although the top and middle layers are heavier than the bottom layer, there were no limitations on the hook's maximum reach for the top and middle layers because their load path does not approach the runway girder.

Based on a review of the licensee's analytical methodology, loads and load combinations, and calculation results, the NRC found that for the actual lifted loads, within the constraints of the prescribed path and crane allowable reach, as discussed above, all members of the RB superstructure would be within the UFSAR allowable stress limits during a postulated OBE event.

<u>Analysis</u>:

The finding that the licensee had not evaluated the effects of live load on the RB crane in combination with seismic loads had the potential to affect the barrier integrity and mitigating systems cornerstones. The finding is of more than minor significance because it affects the cornerstone attribute of design control as it relates to both the Mitigating System and Barrier Integrity cornerstone objectives. In response to the NRC's RAI, the licensee provided an analysis that supported the qualification of the RB superstructure and provided a basis for past operability. Because the RB superstructure was determined to be operable, the finding was of very low risk significance (Green) as described in the introduction to this section.

<u>Enforcement</u>:

Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure applicable regulatory requirements and design bases, as specified in the license, are correctly translated into specifications, drawings, procedures, and instructions. Section 3.8.4.1.3 of the UFSAR required the live load to be included in calculations for Class 1 structures (including the RB) for OBE and SSE load combinations. The licensee's failure to incorporate the rated live load of the RB crane into the original RB OBE load combination calculations from initial operation of Unit 2 on August 11, 1970, to the date of the revised calculation, DRE 98-0020, Revision 1, dated April 23, 2002, is considered a violation of 10 CFR Part 50, Appendix B, Criterion III. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (**NCV 50-237;249/03-002-08**) consistent with Section VI.A.1 of the NRC Enforcement Policy. This issue is documented in the licensee's corrective action program in Condition Report (CR) number 99496.

<u>Description - Crane Bridge and Trolley</u>

The NRC's guidelines for the control and handling of heavy loads state that the equipment which bears load should be capable of withstanding seismic events with rated load on the crane. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants", Section 2.5, "Seismic Design," states that, "... the crane bridge and trolley should be designed and constructed in accordance with Regulatory Guide 1.29, 'Seismic Design Classification,' such that the maximum critical load plus operational and seismically induced pendulum and swinging load effects on the crane should be considered in the design of the trolley, and they should be added to the trolley weight for design of the bridge." Accordingly, licensees are expected to design and construct the lifting system so that an SSE and OBE may not result in any failures that could reduce the functioning of the spent fuel pool storage structure to an unacceptable safety level. Branch Technical Position APCSB 9-1 has similar requirements.

Specifically, BTP APCSB 9-1 states that, "... the crane should be classified as seismic Category I and should be capable of retaining the maximum design load during a safe shutdown earthquake, although the crane may not be operable after the seismic event. The bridge and trolley should be provided with means for preventing them from leaving their runways with or without the design-rated load during operation under seismic loadings. The design-rated load plus operational and seismically-induced pendulum and swinging load effects on the crane should be considered in the design of the trolley, and they should be added to the trolley weight for design of the bridge."

In Special Report 41 (which predated BTP APCSB 9-1), Section 3.2, "Component Failure Analysis," the licensee committed to analyze the crane under the American Institute of Steel Construction (AISC) code requirements for OBE and SSE conditions. In addition, the licensee committed to install lugs or other mechanisms to preclude the trolley and bridge from leaving the runways during a seismic event.

The licensee submitted Supplement A to Special Report 41 by letter dated June 10, 1975, in response to BTP APCSB 9-1. In Supplement A, the licensee stated, "We have reviewed the Branch Position on overhead crane handling systems, dated January 10, 1975, [BTP APCSB 9-1] in light of the system proposed for installation by Commonwealth Edison at Dresden Station. The Dresden and Quad Cities cranes are identified as Safety Class II equipment in the plant operating license. It is not practicable to consider reclassifying the hoist system as Seismic Class I, because this would most probably require a new bridge and extensive modifications to the bridge trackway. The bridge and trolley will be analyzed in a manner consistent with the design codes applicable at the time of original construction, that is, the allowable stress will be limited to 90 percent of yield, with only static lifted loads considered ...."

In 1975-1976, the NRC staff reviewed Supplement A to Special Report 41 against the criteria of BTP APCSB 9-1. The staff issued a safety evaluation report (SER) dated June 3, 1976, approving changes to Dresden, Units 2 and 3, Technical Specifications (TS) governing the operation and surveillance of the upgraded crane with "single-failure-proof" capability. In the SER, the staff stated that the RB crane met the intent of the requirements in BTP APCSB 9-1 for handling casks weighing up to 100 tons with three exceptions: (1) the redundant mechanical limit switch in the main hoist power circuit (for two-blocking), (2) an electrical interlock system to prevent crane travel outside its safe load path, and (3) a slow speed drive motor to limit the hoisting speed. As a result, the handling system was to be operated on a temporary basis, until August 29, 1976, without the installed components, provided that handling operations followed the TS as modified. The staff, in its June 1976 SER, stated, "Based on our review of data provided by the licensee, we have concluded that the integrated design of the crane, controls, and cask lifting devices meets the intent of BTP APCSB 9-1 as regards single failure criteria except in the specific areas of the crane reeving system, and protection against 'two-blocking'." The staff expected the licensee to complete the crane modifications to support continued use of the crane after August 29, 1976, as a single-failure-proof crane. During its review of this issue in 2002, the NRC was unable to determine if the licensee's commitment to analyze the crane bridge and trolley in accordance with applicable codes at the time of the original crane installation, with only static lifted loads considered, was completed, and if the analysis bounded the seismic criteria. Consequently, the NRC issued the RAI to the licensee dated February 26, 2002.

In the RAI, the staff requested the analysis of record that supported the seismic qualification of the bridge and trolley and the load-bearing components of the RB crane. The staff was particularly interested in the design codes and standards used in the analysis to address the seismic qualification of the crane as single-failure-proof. Justification was requested to demonstrate how the crane and supporting structure met the intent of BTP 9-1, including how the analysis addressed SSE plus maximum critical load (lifted load) and how the analysis determined the bridge and trolley would not leave their respective trackways.

The licensee's response to the staff's RAI dated April 12, 2002, reported that the presence of safety lugs on the bridge and trolley had been verified during a walkdown of the RB crane. These safety lugs will ensure the bridge and trolley stay on their respective runways with or without the design load during operation or under seismic loadings. Therefore, the presence of the safety lugs on both the bridge and trolley meet the guidelines of the performance specification in paragraph 3.c of BTP 9-1. The licensee's response to the staff RAI also included an October 10, 1974, crane bridge calculation that verified that OBE and SSE considerations were included in the design of the crane bridge. This satisfied the requirements of BTP APCSB 9-1 for the bridge. However, neither a seismic nor a component failure analysis was available for the crane trolley. Therefore, the licensee could not produce the results of any calculations in support of the crane trolley.

Notwithstanding the lack of any crane trolley calculation, the licensee's response to the RAI included a Revision 1 to calculation DRE 98-0020, which analyzed the RB superstructure with the crane loaded, including pendulum effects, for forces imposed from both an OBE and SSE. The OBE and SSE analysis, with crane-lifted loads up to the current licensing basis of 110 tons, showed that the RB superstructure can support the crane-lifted load with restrictions on the limits of crane travel (i.e., load paths). Since the trolley is located on top of the crane bridge, it will be subjected to similar seismic loading. Because the bridge has been qualified for OBE and SSE conditions, and calculation DRE 98-0020 (Revision 1) demonstrated that the RB superstructure will support the crane, with lifted loads exceeding its licensing basis (125 tons with load travel restrictions), during an OBE, there is reasonable assurance that the crane trolley has been adequately designed to meet the intent of BTP APCSB 9-1.

Therefore, while the NRC was not able to identify the specific technical bases for the NRC conclusion in its 1976 SER that the Dresden crane met the intent of BTP APCSB 9-1, the NRC has determined that the licensee has now adequately demonstrated through analysis, the seismic design qualification of the crane, including lifted loads, as single-failure-proof to 110 tons, and found the crane bridge and trolley to be acceptable. Based on this, the NRC concluded that the licensee has demonstrated that the overhead handling system is capable of withstanding seismic events as a 110 ton single-failure-proof crane within the crane travel limits identified in calculation DRE 98-0020 (Revision 1).

Subsequently, while performing a design basis reconstitution effort on the crane to support a licensing amendment request to increase the single-failure-proof pedigree of the crane to 125 tons, the licensee discovered that a number of components in the trolley did not meet the allowable stresses defined in Crane Manufacturers Association of America (CMAA) Specification 70, 1971 edition. The CMAA specification required that all members be designed with minimum factors of safety of 5 to the ultimate strength of the material. Based on the Whiting Corporation calculation, dated January 23, 2003, the stresses in the trolley and associated hoist components exceeded the CMAA allowable limits by up to 92 percent. This licensee-identified issue is documented in the licensee's corrective action program in Condition Report (CR) number 141038. Although not meeting CMAA allowable limits, all components remained below the yield stress limit. Therefore, this issue is considered to be of minor safety significance. The Whiting Corporation subsequently issued a 10 CFR Part 21 notification regarding the acceptability of RB crane hoist parts to other users in the industry.

Unresolved Item (URI) 07200037/2001-002-05 is closed.

.2 Closed - Unresolved Item (URI 07200037/2001-002-06): Acceptability of licensee characterization of overstress conditions greater than 1.0, but less than 1.1 as "generally acceptable." This item was unresolved pending NRC acceptability of the licensee's characterization, and actual application in practice, of allowing overstress conditions up to 10 percent in seismic analysis as an acceptable "industry practice". The NRC reviewed the licensing basis for the RB superstructure and determined that stresses in the affected members are required to be kept within allowable limits. This was communicated to the licensee during conference calls between the licensee, Region III staff, and NRR staff, including a call on September 13, 2002.

Based upon the discussion for unresolved item 07200037/2001-002-05, all structural members must have ICs less than 1.0 for the specified path of crane travel, to meet requirements specified in UFSAR Section 3.8.4.1.4. Following Revision 1, dated April 23, 2002, Calculation DRE 98-0020, Page 880, still showed roof truss members (under normal snow loading), and interior building column members (under SSE loading) of the RB superstructure with ICs over 1.0. Following additional discussions with Region III and NRR, the licensee documented the issue in the corrective action program under CR number 153412, and issued Revision 2A to Calculation DRE 98-0020, dated April 16, 2003. On April 18, 2003, the licensee conducted another review of the DRE 98-0020, Revision 1, calculation in which an IC of 1.006 was discussed. The licensee concluded that the IC identified as above 1.0, could be reduced to below 1.0 by minimizing the travel of the RB crane hook. The licensee then concluded that all the ICs for the reactor building structural members that were identified by the NRC as above 1.0, could be shown to be below 1.0.

Analysis

This finding had the potential to affect the barrier integrity and mitigating systems cornerstones. The finding is of more than minor significance because it affected the cornerstone attribute of design control as it relates to both the Mitigating System and Barrier Integrity cornerstone objectives. Licensee calculation DRE 98-0020, Revision 1, Page 880, dated April 23, 2002, did not support the qualification of the RB superstructure for a safe shutdown earthquake event for normal snow loading, in that, selected roof truss members and interior building column members would have ICs that exceeded 1.0 for these specific load combinations. Per the licensee's April 23, 2002, calculation, the interior building column members would have been overstressed above minimum yield point in the event of a design basis earthquake. Similarly, the roof truss members would have been overstressed above allowable values under design snow loading, but not above minimum yield point. The risk significance for this potential condition was very low (Green) as described in the introduction to this section. However, despite clear communication to the licensee by the NRC of the requirement pertaining to allowable stresses, i.e., it is not acceptable to rely on overstress of up to 10 percent, it took further NRC intervention and a full year before the licensee adequately resolved the overstress condition of certain structural members identified from the April 2002, 98020 (Revision 1) calculation results.

Enforcement

Title 10 of the Code of the Federal Regulations, Part 50, Appendix B, Criteria III, "Design Control," required, in part, that measures shall be established to assure applicable regulatory requirements and design bases, as specified in the license, are correctly translated into specifications, drawings, procedures, and instructions. Section 3.8.4.1.4 of the UFSAR indicated that stresses, in the case of an SSE, are limited to minimum yield point as a general case, unless a Limit-Design approach is used to determine energy absorption capacity. For other loading conditions, Section 3.8.4.1.4 refers to UFSAR Table 3.8-11, which references maximum allowable stresses for Class 1 structures. The licensee's failure to ensure design stresses in members of the RB superstructure, from initial operation of Unit 2 on August 11, 1970, until April 18, 2003, were limited to allowable design stresses, is considered a violation of 10 CFR Part 50, Appendix B, Criteria III. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (**NCV 50-237;249/03-002-09**) consistent with Section VI.A.1 of the NRC Enforcement Policy.

Unresolved Item (URI) 07200037/2001-002-06 is closed.

.3 Closed - Unresolved Item (URI 07200037/2001-002-07): Adequacy of licensee verification of operability of crane overload protection associated with “new load indication system.” This item was unresolved pending NRC acceptability of the licensee’s demonstration of the proper performance (calibration) of the new load cell and associated overload protection trip feature.

The original licensing basis for the RB crane as a single-failure-proof crane incorporated reliance on an overload protection feature as described in the UFSAR Section 9.1.4.2.2. A digital-type weight indicator for the main hoist is provided such that when the weight to be lifted is above the setpoint on the weight indicator, the control circuit will prevent hoist operation. The new digital load cell system installed in early 2001 replaced the original system that had previously been rendered inoperable. Based on the results of subsequent NRC inspections, the NRC staff determined that the new digital system was not functioning properly. Load cell inaccuracies of up to 40 percent were identified.

After additional work on the load cell system to make it more reliable, the licensee conducted additional load cell calibrations from September to October 18, 2001. A calibrated 300,000 pound capacity dynanometer was used to weigh the reactor shield blocks. The readings on the load cell were within ±2 percent of the dynanometer, which was well within acceptable standards. This calibration surveillance, PM RQ 607 and 608, has now been entered into the Dresden Predefined Parameters computer database and is required to be performed before each scheduled refueling outage for each unit. This finding was considered to be of minor safety significance because the inaccuracies identified by the inspector were in the conservative direction and the load cell would likely have performed the required trip function prior to the overload capacity of the crane being reached. This issue is documented in the licensee’s corrective action program in CR number 153419. This is another example of where substantive NRC involvement was needed to satisfactorily resolve this issue.

Unresolved Item (URI) 07200037/2001-002-07 is closed.

40A6 Meetings

The inspectors presented the inspection results for the integrated report to Mr. R. Hovey and other members of the licensee staff on April 8, 2003. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

Interim Exit Meetings

Safeguards Inspection with Ms. V. Gengler on February 27, 2003.
Unit 2/3 Crane Issues Inspection with Mr. D. Bost on April 24, 2003.

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-237/03-002-01	NCV	Maintenance Workers Perform Unauthorized Work on 2B Containment Cooling Service Water Pump
50-237/03-002-02 50-249/03-002-02	URI	Adequacy of Site Welding Program
50-237/03-002-03 50-249/03-002-03	NCV	Station's Untimely Response to Generic Non-conforming Condition with 480 Volt Motor Control Center Auxiliary Contact Assemblies
50-237/03-002-04 50-249/03-002-04	FIN	Inadequate Operability Evaluation Prepared for Generic Non-conforming Condition if 480 Volt Motor Control Center Auxiliary Contact Assemblies
50-249/03-002-05	NCV	Inadequate Corrective Actions for Exceeding MELLLA Limit
50-249/03-002-06	NCV	Unexpected Half Scram On Unit 3 During Fuse Inspection
50-249/03-002-07	FIN	Unit 3 Scram on Loss of Main Shaft Oil Pump
50-237/03-002-08 50-249/03-002-08	NCV	Violation of 10 CFR Part 50, Appendix B, Criterion III, Including Live Load in the Reactor Building Crane Calculation.
50-237/03-002-09 50-249/03-002-09	NCV	Violation of 10 CFR Part 50, Appendix B, Criterion III, Overstressed Structural Steel

Closed

50-237/03-002-01	NCV	Maintenance Workers Perform Unauthorized Work on 2B Containment Cooling Service Water Pump
50-237/03-002-03 50-249/03-002-03	NCV	Station's Untimely Response to Generic Non-conforming Condition with 480 Volt Motor Control Center Auxiliary Contact Assemblies

50-237/03-002-04 50-249/03-002-04	FIN	Inadequate Operability Evaluation Prepared for Generic Non-conforming Condition if 480 Volt Motor Control Center Auxiliary Contact Assemblies
50-249/03-002-05	NCV	Inadequate Corrective Actions for Exceeding MELLLA Limit
50-249/03-002-06	NCV	Unexpected Half Scram On Unit 3 During Fuse Inspection
50-249/03-002-07	FIN	Unit 3 Scram on Loss of Main Shaft Oil Pump
50-249/2002-002	LER	Reactor Scram Due to Main Shaft Oil Pump Failure
50-237/03-002-08 50-249/03-002-08	NCV	Violation of 10 CFR Part 50, Appendix B, Criterion III, Including Live Load in the Reactor Building Crane Calculation
50-237/03-002-09 50-249/03-002-09	NCV	Violation of 10 CFR Part 50, Appendix B, Criterion III, Overstressed Structural Steel
72-037/2001-002-05	URI	Long-term acceptability of the Unit 2/3 Reactor Building, the reactor building crane, and ancillary equipment for handling large numbers of dry fuel storage casks.
72-037/2001-002-06	URI	Acceptability of licensee characterization of overstress conditions greater than 1.0, but less than 1.1 as “generally acceptable.”
72-037/2001-002-07	URI	Adequacy of licensee verification of operability of crane overload protection associated with “new load indication system.”

Discussed

None

KEY POINTS OF CONTACT

Licensee

D. Bost, Station Director
H. Bush, Lead Radiation Protection Supervisor
J. Ellis, Performance Monitoring Group Lead
T. Fisk, Chemistry Manager
R. Gadbois, Shift Operations Superintendent
V. Gengler, Dresden Site Security Director
J. Hansen, Regulatory Assurance Manager
J. Henry, Operations Director
R. Hovey, Site Vice President
T. Loch, Supervisor, Turbine Systems Group
J. Reda, Design Engineer
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
S. Taylor, Radiation Protection Director

NRC

M. Ring, Chief, Division of Reactor Projects, Branch 1
C. Miller, Chief, Division of Nuclear Materials Safety, Decommissioning Branch
D. Smith, Dresden Senior Resident Inspector
B. Dickson, Dresden Resident Inspector
P. Pelke, Reactor Engineer
B. Lerch, Project Engineer
R. Landsman, Project Engineer, DNMS

IDNS

R. Zuffa, Resident Inspector Section Head, Illinois Department of Nuclear Safety
R. Schulz, Illinois Department of Nuclear Safety

LIST OF ACRONYMS USED

AOV	Air Operated Valve
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	Condition Report
DIS	Dresden Instrument Surveillance
FIN	Finding
HPCI	High Pressure Coolant Injection
IDNS	Illinois Department of Nuclear Safety
IP	Inspection Procedure
IMC	Inspection Manual Chapter
LER	Licensee Event Report
MOV	Motor Operated Valve
MSOP	Main Shaft Oil Pump
MWe	megawatts electrical
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OA	Other Activities
OE	Operability Evaluation
RPV	Reactor Pressure Vessel
SDP	Significance Determination Process
TIL	Technical Information Letter
TS	Technical Specifications
URI	Unresolved Item
WO	Work Order

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignments

CR#141227; Emergency maintenance department personnel placed unit 3 alternate battery of equalize rather than unit 3; dated January 27, 2003

WO#529815-01; Remaining jumpers from unit 2 250 Vdc battery cell #10 and replacement of degraded cell

1R05 Fire Protection

CR#139268; Degraded fire barrier in penetration F-143-04; dated January 13, 2003

CR#139083; XL3 zone 43 devices 1 - 16; dated January 11, 2003

CR#139074; Inoperable fire protection detector; dated January 11, 2003

CR#137946; Missed M2 type fire barriers during initial review; dated January 02, 2003

CR#145241; Transco type M-2 fire barriers found with <8" fire blanket; dated February 19, 2003

1R11 Licensed Operator Requalification

CR#149013; Team 5 clock reset due to out of the box exam failures; dated March 13, 2003

1R13 Maintenance Risk Assessments and Emergent Work Control

CR#139445; 2B containment cooling service water pump; dated January 14, 2003

WO# 99257632-01; Safety related contact testing emergency diesel generator output breaker

SWR# 85597; Unit 2 emergency diesel generator trip on low cooling water pressure when speed lowered

WO# 538688-01; pump & piping pressure testing

CR#144497; Unit 2 emergency diesel generator tripped during maintenance run; dated February 13, 2003

CR#144498; Unit 2 emergency diesel generator tripped on low coolant pressure due to mis-communication; dated February 13, 2003

WO#99118705-04; installation of HPCI are temperature switch

WO#99137212-01; replacement of auxiliary oil pump control switch

WO#522075-04; and modification of MO 2-3301-8 opening logic and performance test (SP03-01-001)

WO#99052602 and 990520402; 2/3 emergency diesel generator oil transfer pump maintenance and surveillance

CR#151092; 2/3 emergency diesel generator output breaker to Bus 23-1 “tripped free” when given a signal to close during MA-DR-773-304

1R15 Operability Evaluations

CR#138662; D2 250 Vdc battery (potential leak on battery cell 10); dated January 08, 2003

CR#137884; Missing bolts on cable tray base support to concrete floor; dated December 30, 2002

CR#137334; General Electric notified that control rod blades have been manufactured with suspect tube; dated December 23, 2002

CR#139819; TEC breaker setting; dated January 16, 2003

CR#144138; Damper would not fully open; dated February 12, 2003

CR#144887; Operability evaluation expectations not met; dated February 19, 2003

CR#144701; Operability scope not fully defined; dated February 14, 2003

CR#148567; Nuclear Oversight notes Part 21 RVW not performed / some maintenance concerns; dated March 11, 2003

CR#146326; Regarding alignment of suction piping to the unite 2 diesel generator cooling water pump. A misalignment of 3/4 exists, vendor specification is that there should be no than ½ inch misalignment; dated February 26, 2003

1R19 Post Maintenance Testing

CR#139085; Misposition of control rod drive F-2 due to mechanical failure; dated January 11, 2003

CR#139059; Unit 2 control rod drive H-6 scram times slow at multiple insert positions; dated January 11, 2003

CR#139053; Scram valves for control rod drive M-12 slow to reset; dated January 11, 2003

CR#142226; Required post maintenance test not performed; dated January 30, 2003

DOS 1600-04; Revision 18, “U2/3 Quarterly Valve Timing”

DOS 1600-03; Revision 29, “Quarterly Valve Timing”

DOS 1500-01; Revision 25, “Low Pressure Coolant Injection System Valve Operability and Timing”

WR# 99100074-03; Added packing to air operator valve 2/3-5741-48B, service water supply to control room air conditioning unit B

WR# 00542977-01; Repair aux contacts unit 2 low pressure coolant injection LOOP I and II crosstie motor operated valve

WR# 00542868-01; Repair aux contacts unit 2 isolation condenser reactor inlet isolation valve

CR#146057; MO 2-1501-22A fails post maintenance test after maintenance; dated February 24, 2003

CR#150204; Unit 3 high pressure coolant injection system auxiliary cooling water pump; dated March 21, 2003

1R22 Surveillance Test

CR#141227; Unit 3 ALT batteries put on equalize instead of unit 3 Main battery; dated January 27, 2003

CR#143991; Technical Specification 2-6641-528 found full of oil-required replacement; dated February 11, 2003

CR#143995; Technical Specification 2-6641-524 failed microswitches; February 11, 2003

CR#143153; DIS 1300-07, “Isolation Condenser Hi Flow Monthly Calibration;” dated February 7, 2003

CR# 143007; 3-0203-3D Pressure controller switch found out of calibration; dated February 6, 2003

CR#142692; Inadequate technical specification surveillance; dated February 6, 2003

DIS 0263-02; Reactor Vessel Wide Range Level Transmitter LT 2/3 Calibration and Maintenance Inspection, Revision 7

DOS 0500-04; “Turbine Control Valve Fast Closure (load reject) Scram Circuit Function Test,”, Revision 9

DIS 1500-04; “Containment Spray Interlock, Containment High Pressure Switches Calibration,” Revision 4

DOS 1400-02, “Core Spray System Pump Test with Torus Available,” Revision 27

WO# 99228573-01

CR#146091; D2 IMD anticipated transient without scram procedure ‘as found’ tolerance exceed technical specification value; dated February 25, 2003

CR#146520; Technical specification instrument channel check deficiencies; dated February 27, 2003

CR#146625; #4 turbine control valve failed to operate properly; dated February 27, 2003

CR#148957; 2-2301-31 Automatic open valve stroke time in the IST alert range; dated March 13, 2003

CR#148389; Unit 3 diesel generator cooling water pump IST flow rate trend adverse; dated March 10, 2003

CR#148329; Technical Specification instrument channel calibration deficiencies; dated March 10, 2003

CR#148326; D3 containment cooling service water pump vault penetration seals fail as-found leak test; dated March 5, 2003.

CR#147457; Nuclear oversight report the floor drain in control rod drive pump floor was leaking. Containment cooling service water sub door leak rate in progress.; dated Marcy 4, 2003

CR#146485; Received unexpected ½ scram channel “b” after performing load reject scram circuit functional test for #3 control valve; dated February 27, 2003

WO#521240-01; Turbine fast closure (load reject) scram circuit functional test

1R23 Temporary Modification (71111.23)

CR#142921; Temporary configuration change extension; dated February 6, 2003

CR#148077; Insufficient temporary modification accelerometer location data provided; March 7, 2003

CR# 138662; Cracked cell - unit 2 250Vdc battery rack; dated January 8, 2003

3PP4 Physical - Security Plan Changes (71130.04)

Dresden Nuclear Power Station Plan; Revision 68; dated February 2003

Dresden Nuclear Power Station Safeguards Contingency Plan, Revision 68; dated February 2003

71151 Performance Indicator Verification

CR#142738; Potential white finding for high pressure coolant injection system pipe support damage; dated February 6, 2003

71152 Problem Identification and Resolution

CR#138515; High FCL following load drop; dated January 8, 2003

CR#138247; Unit 3 emergency diesel generator airbox support foot found cracked; dated January 6, 2003

CR#144917; Nuclear Oversight identifies nonconformance not addressed in a timely and correct; dated February 20, 2003

CR#148115; Corrective action program coordinator identifies untimely processing of two condition reports; March 4, 2003

CR#148567; Nuclear Oversight notes Part 21 RVW not performed / some maintenance concerns; March 11, 2003

CR#148605; CAPR-2 of RCR 92415-02 found ineffective; dated March 11, 2003

CR#148890; Condition reports not generated for maximum extended load line limit analysis boundary violation in July; dated March 12, 2003

71153 Event Follow-up

CR#141787; No information in previous events section of Licensee Event Report; dated January 28, 2003

CR 116478; Unit 3 reactor scram due to turbine trip; dated July 21, 2003

CR 115691; Loss of main turbine speed indication/permanent magnet generator failure; dated July 15, 2003

DES 5600-05; Revisions 2 and 3, “Turbine Shaft Voltage Surveillance”

4OA5 Other

CR 99496; Reactor Building Superstructure Seismic Evaluation, dated March 15, 2002

CR 141038; Rx Bldg Crane Main Hoist Components Do Not Meet CMAA F.O.S, dated January 24, 2003.

CR 153412; NRC URI 2001-002-06: Overstress in RB Superstructure Girder, dated Apri 10, 2003.

CR 153415; NRC Potential Finding - NUREG 0612 Safe Load Path Markings, dated April 10, 2003

CR 153419; NRC URI 2001-002-07: Crane Load Cell Testing, dated April 10, 2003