UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

August 2, 2005

Randall K. Edington, Vice
President-Nuclear and CNO
Nebraska Public Power District
P.O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000298/2005003

Dear Mr. Edington:

On June 23, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Cooper Nuclear Station. The enclosed integrated inspection report documents the inspection findings which were discussed on July 7, 2005, with Mr. S. Minahan, General Manager of Plant Operations, and other members of your staff.

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

Based on the results of this inspection, the NRC identified six findings which were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC also determined that there were three violations associated with these findings. However, because these violations were of very low safety significance and the issues were entered into the licensee's corrective action program, the NRC is treating these findings as noncited violations, consistent with Section VI.A.1 of the NRC's Enforcement Policy. These noncited violations are described in the subject inspection report. If you contest the violations or significance of the violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper Nuclear Station facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records 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).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Wayne C. Walker, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-298
License: DPR-46

Enclosure:
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w/attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket.:	50-298
License:	DPR-46
Report:	05000298/2005003
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	March 25 through June 23, 2005
Inspectors:	S. Schwind, Senior Resident Inspector S. Cochrum, Resident Inspector L. Ricketson, Senior Health Physicist K. Clayton, Operations Engineer S. Garchow, Operations Engineer L. Vick, Reactor Engineer Paul J. Elkmann, Emergency Preparedness Inspector
Approved By:	W. Walker, Branch C, Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000298/2005003; 03/25/05 - 06/23/05; Cooper Nuclear Station, Personnel Performance During Nonroutine Evolutions, Access Control to Radiological Significant Areas, Problem Identification and Resolution and Other Activities.

The report covered a 3-month period of inspection by resident inspectors and region-based inspectors. Three Green noncited violations and three Green findings were identified. The significance of the issues is indicated by their color (Green, White, Yellow, or Red) and was determined by the significance determination process in Inspection Manual Chapter 0609. Findings for which the significance determination process does not apply are indicated by the severity level of the applicable violation. 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. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, was identified regarding inadequate corrective actions which resulted in the high pressure coolant injection system being rendered inoperable during scram recovery actions on April 15, 2005. During the scram recovery, operators disabled the system by placing the auxiliary oil pump in pull-to-lock rather then aligning the system to a standby condition as required by procedures. This was the third occurrence of this error in 2 years.

  This finding was more than minor since it affected the availability of the high pressure coolant injection system which is relied upon to mitigate the consequences of an initiating event. Based on the Significance Determination Process Phase 1 screening, this finding was determined to have very low safety significance since it did not represent the actual loss of a safety function for greater than its Technical Specification allowed outage time and did not screen as risk significant due to external initiating events. This finding also had crosscutting aspects associated with problem identification and resolution since this was the third occurrence of this event and previous corrective actions were not comprehensive in addressing the causes. In addition, the condition report documenting this issue was incorrectly classified in the corrective action program until questioned by the inspectors. The licensee entered this finding into their corrective action program as CR-CNS-2005-02982 (Section 1R14).

- Green. The inspectors identified a noncited violation of Appendix B, Criterion XVI of 10 CFR Part 50, for failure to take adequate corrective actions for degraded conditions on the service water booster pump system. On April 5, 2005, water intrusion into the service water Booster Pump A outboard bearing oil rendered the pump inoperable. This was the second occurrence.

This finding was considered more than minor since it affected the operability, availability, and reliability of a mitigating system. It was considered to have very low safety significance, since it did not represent the actual loss of a safety function. It also had crosscutting aspects associated with problem identification and resolution since the previous corrective actions only addressed the symptoms of the adverse condition, not the root cause. The licensee entered this finding into their corrective action program as CR-CNS-2005-02732 (Section 4OA2).

- Green. A self-revealing finding was identified involving the failure to perform an adequate design change for the reactor feed system startup flow control valves. The inadequate design change failed to ensure component temperature ratings were not exceeded, which would adversely affect valve operation. Specifically, the licensee's evaluation failed to recognize and address acceptable O-ring types for the temperatures of the reactor feed system.

  This finding is greater than minor because it affected the cornerstone attribute of design control. It was determined to have very low safety significance in a Phase 3 evaluation. This finding has crosscutting aspects associated with human performance based on the fact that engineering did not follow appropriate guidance in evaluating system environmental conditions related to installing the modification. The licensee entered this finding into their corrective action program as CR-CNS-2004-06997 (Section 4OA5).

Cornerstone: Barrier Integrity

- Green. A noncited violation of License Condition 2.C(1) occurred when operators allowed reactor power to exceed the licensed power limit of 2381 MW for 7 hours during a xenon transient on April 10, 2005. Reactor power slowly increased above 2381 MW during the transient; however, operators were controlling the reactor using the eight hour power average which remained below 2381 MW for approximately 7 hours. Reactor power remained below 102 percent during the entire transient; therefore, the reactor was not operated outside its design limits.

  This finding was more than minor since it affected the cornerstone attribute of maintaining functionality of the fuel cladding. Based on the Significance Determination Process Phase 1 screening, this finding was determined to have very low safety significance since it only involved the potential to affect the fuel barrier. This finding also had crosscutting aspects associated with human performance and problem identification and resolution since the cause of this event was the erroneous belief by the reactor operator that the reactor could be operated above licensed thermal power as long as the 8-hour average remained below the licensed limit. This aspect of the event was not addressed in the licensee's apparent cause. The licensee entered this finding into their corrective action program as CR-CNS-2005-02869 (Section 1R14).

Cornerstone: Occupational Radiation Safety

- Green. The inspector identified a finding because the licensee failed to plan and control dose or provide ALARA Committee oversight for the work activity that accrued the largest portion of the refueling outage dose. The drywell general access and limited maintenance special work permit accrued nearly 38 person-rem, but had no dose estimate, work plan, or ALARA committee review.

  This finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute (ALARA planning/estimated dose) and affected the associated cornerstone objective in that the failure to plan and control radiation dose affected the licensee's ability to ensure adequate protection of worker health and safety. In this case, the licensee formulated no dose estimate. Manual Chapter 0308, Appendix C, states, "Planned or intended collective dose can be the results of a realistic dose estimate (or projection) established during ALARA planning or the dose expected by the licensee (i.e., historically achievable) for the reasonable exposure control measures specified in ALARA procedures/planning." Since the licensee had no expectation of the potential dose, the inspector compared the actual dose with historical doses and found that the 2005 doses exceeded the historical totals by more than 50 percent. When processed through the Occupational Radiation Safety Significance Determination Process, this ALARA finding was found to have no more than very low safety significance because the finding was related to ALARA, but the licensee's 3-year rolling average collective dose was not greater than 240 person-rem. The finding was documented in the licensee's corrective action program as CR-CNS-2005-2985 (Section 2OS2).

- Green. The inspector identified a finding because inadequate planning resulted in the collective dose of a work activity that exceeded 5 person-rem and exceeded the dose estimate by more than 50 percent. Radiological Job Package 2005AL-03, Sludge Removal from the Torus, was projected to accrue 3.2 person-rem, but actually accrued approximately 5.7 person-rem because inadequate planning necessitated additional, unplanned handling of radioactive filters.

  This finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute (ALARA planning/estimated dose) and affected the associated cornerstone objective in that the failure to control collective dose affected the licensee's ability to ensure adequate protection of the worker health and safety from exposure to radiation. When processed through the Occupational Radiation Safety Significance Determination Process, this ALARA finding was of very low safety significance because the finding was related to ALARA, but the licensee's 3-year rolling average collective dose was not greater than 240 person-rem. The finding was documented in the licensee's corrective action program as CR-CNS-2005-2969 (Section 2OS2).

REPORT DETAILS

The plant was operating at full power at the beginning of this inspection period. On April 15, 2005, an automatic reactor scram occurred due to a failure in the master reactor vessel water level controller. The full power operations resumed on April 18 following repairs. Reactor power was reduced to approximately 70 percent on April 19 for a control rod pattern adjustment. On June 10, reactor power was reduced to approximately 30 percent to replace reactor recirculation motor generator brushes. The reactor was returned to full power on June 12 following repairs.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors selected activities representing the review of preparations for hot weather and adverse weather conditions (two inspection samples):

The inspectors selected four activities representing the review of preparations for hot weather conditions on two risk significant systems (one inspection sample). The four activities included:

- A review of maintenance work orders completed in order to prepare the systems for possible high temperatures
- A review of deficiency tags and condition reports associated with hot weather protection measures to determine their impact on the systems
- A walkdown of the main power transformer yard to determine if ventilation is adequate for warm weather per procedures
- A walkdown of the ventilation screens in the intake structure to verify that the licensee had completed the required actions identified in the work orders

The two systems chosen for this inspection included:

- Portions of the reactor building heating and cooling ventilation system
- The intake structure and environmental controls in the service water pump room

The inspectors also reviewed activities in preparations for adverse weather conditions during a tornado warning on April 21 (one inspection sample). The activities included:

- A review of the control room logs to ensure safety systems were prepared for adverse weather conditions

- A review of deficiency tags and condition reports to determine their impact on systems

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Equipment Alignment Inspections

a. Inspection Scope

The inspectors performed three partial equipment alignment inspections (three inspection samples). The walkdowns verified that the critical portions of the selected systems were correctly aligned per the system operating procedures. The following systems were included in the scope of this inspection:

- Service Water (SW) System Loop B while Loop A was out of service for planned maintenance on April 5. The walkdown included portions of the system in the SW pump room and the control room.

- Reactor Equipment Cooling System Loop A while Loop B was out of service for planned maintenance on May 9. The walkdown included portions of the system in the reactor building and the control room.

- Alternate emergency core cooling injection systems on June 8. The walkdown included portions of the system in the reactor building and turbine building.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

On April 19, the inspectors performed one complete system alignment inspection of the SW system following planned maintenance. The inspectors verified that the system was in the appropriate configuration per the system operating procedure and that it was installed and capable of performing its design functions as described in the Updated Safety Analysis Report. A review of maintenance work orders and corrective action documents for the past 12 months was also performed. A walkdown of the system was performed to assess material conditions, such as system leaks and housekeeping issues, that could adversely affect system operability (one inspection sample).

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed six fire zone walkdowns to determine if the licensee was maintaining those areas in accordance with its fire hazards analysis report (six inspection samples). The walkdowns verified that fire suppression and detection equipment was operable, that transient combustibles and ignition sources were appropriately controlled, and that passive fire protection features were in place and operable as required by the fire hazards analysis report. The following areas were included in the scope of this inspection:

- Fire Zone 1E, High pressure coolant injection room
- Fire Zone 7A, Service water booster pump room
- Fire Zone 6, Refueling floor
- Fire pump house, Fire pump room
- Fire Zone 9B, Cable expansion room
- Fire Zone 2A, Reactor building 903 elevation

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors performed an inspection of the external flood protection features for the emergency diesel generator rooms (one sample). The inspection included a walkdown of flood protection features, a review of procedures, the Updated Final Safety Analysis Report, selected design criteria documents, and design calculations, including:

- Cooper Nuclear Station Design Criteria Document 38, "Internal Flooding System," Revision 2

- Calculation NEDC 91-069, "Moderate Energy Line Break Flooding", dated June 12, 1991

The walkdown verified that flood protection features were in place and operable as required by the flooding analysis for the emergency diesel generator rooms.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Quarterly Inspection

a. Inspection Scope

On March 30, the inspectors observed one session of licensed operator requalification training (one inspection sample) in the plant simulator. The training evaluated operator ability to perform recovery actions for a loss of offsite power sources, reactor scrams, and small break loss of coolant accidents. Observations were focused on the following key attributes of operator performance:

- Crew performance in terms of clarity and formality of communications
- Ability to take timely and appropriate actions
- Prioritizing, interpreting, and verifying alarms
- Correct implementation of procedures, including the alarm response procedures
- Timely control board operation and manipulation, including high-risk operator actions
- Oversight and direction provided by the shift supervisor, including the ability to identify and implement appropriate Technical Specification requirements, reporting, emergency plan actions, and notifications
- Group dynamics involved in crew performance

The inspectors also verified that the simulator response during the training scenario closely modeled expected plant response during an actual event.

b. Findings

No findings of significance were identified.

.2 Biennial Inspection

a. Inspection Scope

The inspectors reviewed two unresolved items documented in NRC Inspection Report 05000298/2004015 dated March 16, 2005, for the licensed operator requalification program. Specifically, this inspection assessed the Cooper Nuclear Station's plant-referenced simulator for compliance with 10 CFR 55.46 using Baseline Inspection Procedure 71111.11 (Sections 02.11, 03.11, and Appendix C). This assessment included: (1) the adequacy of the licensee's simulation facility for use in operator licensing examinations; and (2) the effectiveness of the licensee's process for continued assurance of simulator fidelity with regard to identifying, reporting, correcting, and resolving simulator discrepancies as required by 10 CFR 55.46(d).

Unresolved Items 05000298/2004015-01, "Simulator Fidelity," and 05000298/2004015-02, "Acceptance Criteria for Simulator Performance Testing," are discussed in Section 4OA5. These two unresolved items were reviewed by the inspectors to ascertain if the simulator satisfied the requirements in 10 CFR 55.46 for a plant-referenced simulator. The licensee requested a public meeting, which was held on February 24, 2005. At this meeting, the licensee indicated that the simulator unresolved items documented by the inspection team were primarily the result of a weak simulator discrepancy resolution process and their failure to adequately communicate an accurate status of their simulator discrepancies. During the meeting, the licensee stated that their discrepancy resolution process had been restructured to make it more effective and controlled.

In order to assess the licensee's simulation facility for use in operator licensing examinations, simulator fidelity was evaluated against: (1) the identified 2004 performance tests and malfunction test packages to determine if they satisfied the requirements as stated in American National Standards Institute Standard 3.5, Revision 1, for simulator testing; (2) the open and closed discrepancy lists to evaluate the impact of these on 10 CFR 55.45 and 55.59 operator actions and to verify that the identified closed out deficiencies have been corrected based on package review and scenario testing; and (3) the expected simulator response to specific malfunctions and evolutions identified to determine if these responses were appropriate based on expected plant response to the same initiators. One inspector was assigned to the simulator control room for the entire week to observe scenario validations (identified as dynamic scenario tests) and to evaluate the adequacy of the simulator response. The inspector also created specific scenarios (identified as nonlibrary scenario tests) that would exercise the simulator in areas where the discrepancy reports showed potential fidelity issues. The main focus of these nonlibrary scenario tests was mainly anticipated transient without scram and loss of coolant accident scenarios. The inspector observed 23 scenarios covering more than 100 malfunctions with no major fidelity issues documented.

In order to evaluate the effectiveness of the licensee's process for continued assurance of simulator fidelity with regard to identifying, reporting, correcting, and resolving simulator discrepancies via a corrective action program, several items were reviewed, including: (1) the current revision of the identified procedures used to characterize, classify, disposition, and close simulator discrepancies; and (2) internal memoranda identified from the simulator group to the simulator review committee, which is responsible for the characterization and timely disposition of simulator discrepancies.

The inspectors also interviewed members of the licensee's simulator configuration control group as part of this review.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed two equipment performance issues (two inspection samples) to assess the licensee's implementation of their maintenance rule program. The inspectors verified that components which experienced performance problems were properly included in the scope of the licensee's maintenance rule program and that the appropriate performance criteria were established. Maintenance rule implementation was determined to be adequate if it met the requirements outlined in 10 CFR 50.65 and Administrative Procedure 0.27, "Maintenance Rule Program," Revision 15. The inspectors reviewed the following equipment performance problems:

- Failure of the reactor level controller on April 15 (CR-CNS-2005-02983)
- Failure of SW Booster Pump A on April 5 (CR-CNS-2005-02732)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed five risk assessments (five inspection samples) for planned or emergent maintenance activities to determine if the licensee met the requirements of 10 CFR 50.65(a)(4) for assessing and managing any increase in risk from these activities. Evaluations for the following maintenance activities were included in the scope of this inspection:

- Risk associated with the installation of silt plates in the intake bays on March 31 (Work Order 4432520)
- Risk associated with corrective maintenance on SW Booster Pump A on April 5 (Work Order 4434555)

- Risk associated with a 345 kv switchyard breaker replacement on April 7 (Work Order 4427220)
- Risk associated with the 345 kv-161 kv auto-transformer corrective maintenance on April 27 (Work Order 4431380)
- Risk associated with emergent corrective maintenance on the Main Power Transformer A oil pump on June 11 (Work Order 4445139)

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Evolutions

a. Inspection Scope

For the four nonroutine events described below (four inspection samples), the inspectors reviewed operator logs, plant computer data, and strip charts to determine what occurred, how the operators responded, and whether the response was in accordance with plant procedures.

- The inspectors reviewed control room logs, conducted interviews, and reviewed the licensee’s apparent cause and corrective actions for exceeding their licensed thermal power level on April 10.
- On April 15, the inspectors responded to the control room shortly after the reactor automatically scrammed due to low reactor water level in response to reactor vessel level controller failure causing reactor water level to drop rapidly. During this event, the inspectors monitored plant conditions and observed and evaluated the follow-up actions by the operators and action required by procedures.
- On April 21, the inspectors responded to the control room and observed site response to a tornado warning issued for Nemaha County, NE. The inspectors verified that operator response was in accordance with station procedures and monitored plant conditions during the warning.
- On April 27, the inspectors responded to the control room shortly after a fire was reported in the intake structure. During this event, the inspectors monitored plant conditions and observed and evaluated the actions by the control room and actions required by procedures.

b. Findings

Reactor Operation in Excess of Licensed Thermal Power Limits

Introduction. A Green noncited violation (NCV) of License Condition 2.C(1) occurred when operators allowed the 8-hour power average to exceed 2381 MW during a xenon transient.

Description. On April 8, 2005, operators reduced reactor power to approximately 90 percent in order to restore a control rod to the desired position following a rod operability test during which the rod control system timer malfunctioned. The reactor remained at that power level during repairs to the rod control system until the morning of April 10, when full power operation was resumed. A xenon transient was in progress throughout most of the day on April 10 which caused reactor power to slowly increase. Operators were maintaining the 8-hour power average below the licensed power limit of 2381 MW. Shortly after the day shift crew was relieved, the night shift reactor operator noted that the 8-hour power average was above the licensed power limit. The 8-hour average peaked at 2381.2 MW before actions were taken to reduce reactor power. A subsequent review of power history indicated that the reactor had been operated between 2381 MW and 2382 MW for approximately 7 hours.

This event was evaluated under Condition Report 2005-2869, which stated that the cause for the 8-hour power average exceeding the licensed thermal power limit was the fact that operators "maintained an hourly average above 2381 [MW]." There was no discussion of operator attentiveness, operator training, procedure quality, human performance, or any other factor that may have contributed to this event. Corrective actions included changes to General Operating Procedure 2.1.10, "Station Power Changes," which required the 1-hour power average to be maintained below 2381 MW. In addition, reactor engineering performed an evaluation which determined that no thermal limits were challenged by this event.

Analysis. The failure to maintain steady state reactor power at or below the licensed thermal power limit was considered to be a performance deficiency since it was reasonably within the licensee's ability to do so. This finding affected the Barrier Integrity Cornerstone and was more than minor since it affected the cornerstone attribute of maintaining functionality of the fuel cladding. Based on the Significance Determination Process Phase 1 screening, this finding was determined to have very low safety significance since it only involved the potential to affect the fuel barrier.

This finding also had crosscutting aspects associated with human performance and problem identification and resolution. Based on interviews, the inspectors learned that the true cause of this event was the belief by the reactor operator that the reactor could be operated above licensed thermal power as long as the 8-hour average remained below the licensed limit. Therefore, this was a human error. In addition, this aspect of the event was not addressed in the licensee's apparent cause.

Enforcement. License Condition 2.C(2) of the Cooper Nuclear Station Operating License states that the licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2381 MW (thermal). Contrary to this, on April 10, 2005, the reactor was operated between 2381 and 2382 MW for approximately 7 hours. Because this violation was of very low safety significance and was entered in the corrective action program as CR-CNS-2005-02869, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000298/2005003-01, Reactor Operation in Excess of Licensed Thermal Power Limits.

High Pressure Coolant Injection Rendered Inoperable During Scram Recovery

Introduction. A Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, was identified regarding inadequate corrective actions which resulted in the high pressure coolant injection (HPCI) system being rendered inoperable during scram recovery actions.

Description. On April 15, 2005, an automatic reactor scram occurred due to low reactor vessel water level. The cause of the event was a malfunction of the reactor vessel level master controller which sent an erroneous signal to both reactor feed pumps, causing their speed to decrease to a minimum value. Immediately following the scram, HPCI automatically initiated on low vessel level; however, operators quickly took manual control of the reactor feed pumps and HPCI injection was not required. The HPCI initiation signal cleared within 30 seconds of the scram but, rather than placing HPCI in a standby condition in accordance with System Operating Procedure 2.2.33.1, "High Pressure Coolant System Operations," Revision 19, Attachment 1, the operator tripped the HPCI turbine and placed the auxiliary oil pump in pull-to-lock, thereby precluding an automatic initiation of HPCI and rendering the system inoperable. The operator immediately realized this mistake, appropriately notified his supervisor, and restored HPCI to a standby condition. HPCI was inoperable for approximately 20 seconds.

NRC Integrated Inspection Report 05000298/2003-006 documented a Green noncited violation of Technical Specification 5.4.1 for failure to follow procedures during scram recovery actions on May 28, 2003. During this scram, operators placed the HPCI auxiliary oil pump in pull-to-lock rather than placing it in a standby condition. As a corrective action, the licensee determined that the operating procedure needed to be revised; however, these changes were never implemented. NRC Integrated Inspection Report 05000298/2004-002 documented an additional Green noncited violation of Technical Specification 5.4.1 since operators performed the same error during scram recovery actions on November 28, 2003. In addition, this report documented a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, since the corrective actions for the first event were never implemented. As a result of this finding, corrective actions to revise the procedure were implemented on October 6, 2004. In addition, a training needs analysis was performed for the change which determined that adding the procedure changes to the operators' required reading was adequate training.

The error in securing HPCI on April 15, 2005 was documented in CR-CNS-2005-02982 and a formal root cause analysis was performed. The analysis concluded that securing

HPCI by placing the auxiliary oil pump in pull-to-lock was “long-ingrained” in the operators and adding the procedure changes to the operators’ required reading was insufficient training. Furthermore, this procedure change had not been incorporated into simulator training scenarios as part of the training requirement.

In addition to the immediate corrective actions taken by the operator during the scram recovery, the licensee has further revised System Operating Procedure 2.2.33.1 to clarify operators actions and these changes have been included in simulator training scenarios.

Analysis. The failure to adequately train operators on procedure changes was considered to be a performance deficiency. This finding affected the Mitigating Systems Cornerstone and was more than minor since it affected the availability of the HPCI system, which is relied upon to mitigate the consequences of an initiating event. Based on the Significance Determination Process Phase 1 screening, this finding was determined to have very low safety significance since it did not represent the actual loss of a safety function for greater than its Technical Specification allowed outage time and did not screen as risk significant due to external initiating events.

This finding also had crosscutting aspects associated with problem identification and resolution. This assessment was based on the fact that the licensee had multiple opportunities to correct the system operating procedure and conduct adequate training for operators, but it took three events and 2 years to implement the procedure changes and conduct the required training. In addition, CR-CNS-2005-02982, which documented the operator error on April 15, 2005, was initially classified as a Category C condition report, or “broke-fix”, and was assigned to the operations department. The licensee only upgraded this condition report to a full root cause after the inspectors called attention to Administrative Procedure 0.5.CR, “Condition Report Initiation, Review, and Classification,” Revision 1, which required that this condition report be classified as Category A. Category C condition reports are not required to be evaluated by a multidisciplinary team; therefore, the licensee may not have gained the same insights with respect to operator training had this issue only been evaluated by the operations department rather than a team of individuals which included members from the training department.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, requires that measures be established to assure that conditions adverse to quality are promptly identified and corrected. The actions which rendered HPCI inoperable unnecessarily were considered to be a condition adverse to quality; the corrective actions for this condition included changing the system operating procedure and training operators on the procedure changes. This adverse condition was never fully corrected since operators were not adequately trained on the procedure changes. The failure to correct this condition adverse to quality was considered to be a violation of Criterion XVI; however, because this violation was of very low safety significance and was entered in the corrective action program as CR-CNS-2005-02982, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000298/2005003-02, Inadequate Corrective Actions Result in High Pressure Coolant Injection System being Rendered Inoperable.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed four operability determinations (four inspection samples) associated with mitigating system capabilities to ensure that the licensee properly justified operability and that the component or system remained available so that no unrecognized increase in risk occurred. These reviews considered the technical adequacy of the licensee's evaluation and verified that the licensee considered other degraded conditions and their impact on compensatory measures for the condition being evaluated. The inspectors referenced the Updated Safety Analysis Report, Technical Specifications, and the associated system design criteria documents to determine if operability was justified. The inspectors reviewed the following equipment conditions and associated operability evaluations:

- Temporary radiation shielding installed on the residual heat removal system in the drywell (CR-CNS-2005-03081)
- Failure of a motor termination on Residual Heat Removal Pump A (CR-CNS-2005-03057)
- Startup station service transformer insulator ground (CR-CNS-2005-03633)
- Low ground resistance readings on the emergency station service transformer bus (CR-CNS-2005-03946)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors performed a review of all open operator workaround items (one inspection sample) to evaluate their cumulative effect on mitigating systems and the operator's ability to implement abnormal or emergency procedures. In addition, open operability determinations and selected condition reports were reviewed and operators were interviewed to determine if there were additional degraded or nonconforming conditions that could complicate the operation of plant equipment.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing

a. Inspection Scope

The inspectors reviewed or observed five selected postmaintenance tests (five inspection samples) to verify that the procedures adequately tested the safety function(s) that were affected by maintenance activities on the associated systems. The inspectors also verified that the acceptance criteria were consistent with information in the applicable licensing basis and design basis documents and that the procedures were properly reviewed and approved. Postmaintenance tests for the following maintenance activities were included in the scope of this inspection:

- Corrective maintenance to adjust packing on SW Pump A (Work Order 4401059)
- Planned maintenance on Reactor Equipment Cooling Pump C (Work Order 4354582)
- Corrective maintenance to inspect discoloration on the emergency station service transformer bus insulation (Work Order 4442409)
- Corrective maintenance to replace a failed relay in the rod block monitor (Work Order 4438104)
- Corrective maintenance to replace a leaking fuel injector pump on Emergency Diesel Generator 2 (Work Order 4444359)

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors evaluated the licensee’s outage activities associated with Forced Outage 05-01 to ensure that: risk was considered in developing the outage schedule, administrative risk reduction methodologies were implemented to control plant configuration, mitigation strategies were developed for losses of key safety functions, and the operating license and Technical Specification requirements were satisfied to ensure defense-in-depth. Specifically, the following activities were included in the scope of this inspection:

- Control room observations of the reactor shutdown and startup

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed or reviewed the following five surveillance tests (five inspection samples) to ensure that the systems were capable of performing their safety function and to assess their operational readiness. Specifically, the inspectors verified that the following surveillance tests met Technical Specification requirements, the Updated Safety Analysis Report, and licensee procedural requirements:

- 6.HPCI.103, “HPCI IST and 92 Day Test,” Revision 27, performed on March 30
- 6.LPRM.302, “LPRM Calibration and Setpoint Adjustments,” Revision 9, performed on May 11
- 6.2CS.201, “CS Motor Operated valve Operability Test (DIV 2),” Revision 11, performed on May 11
- 6.1RPS.702, “RPS Primary Containment High Pressure Channel Functional Test (DIV 1),” performed on May 12
- 6.2RPS.702, “RPS Primary Containment High Pressure Channel Functional Test (DIV 2),” performed on May 12

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed one temporary plant modification (one inspection sample), Work Order 4432520, implemented on March 23, which installed flow diverters into circulating water bays. The purpose of this modification was to mitigate the accumulation of sediment into the circulating water bays. The inspectors verified that the change did not require NRC approval prior to implementation, and adequate controls on the installation existed.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert Notification System Testing

a. Inspection Scope

The inspector discussed with licensee staff the status of offsite siren and tone alert radio systems and licensee changes to the siren testing methodology to determine the adequacy of licensee methods for testing the alert and notification system in accordance with 10 CFR Part 50, Appendix E. The licensee's alert and notification system testing program was compared with criteria in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, Federal Emergency Management Agency (FEMA) Report REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants," and the licensee's current FEMA-approved alert and notification system design report. The inspector also reviewed the following procedures:

- Emergency Preparedness Desk Guide 2, Attachment C-1, "Semi-Monthly Alert and Notification System Siren Testing," Revisions 9 through 11

- Emergency Preparedness Desk Guide 2, Attachment G-1, "Issuance of NOAA/EAS Radio Receivers," Revision 0

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing

a. Inspection Scope

The inspector reviewed the following documents related to the emergency response organization augmentation system to determine the licensee's ability to staff emergency response facilities in accordance with the licensee emergency plan and the requirements of 10 CFR Part 50, Appendix E:

- Emergency Preparedness Desk Guide 2, Attachment E-3, "Bi-Monthly ERO Call-In Test," Revision 12

- Evaluations for eight call-in and drive-in drills conducted in 2004 and 2005

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspector reviewed the following documents related to the licensee's corrective action program to determine the licensee's ability to identify and correct problems in accordance with 10 CFR 50.47(b)(14) and 10 CFR Part 50, Appendix E:

- Procedure 0.5, "Conduct of the Condition Report Process," Revision 52
- Procedure 0.5 EVAL, "Preparation of Condition Reports," Revision 8
- CAP Desk Guide 11, "CNS Risk Significance Determination Screening Process," Revision 5
- Summaries of 279 corrective actions assigned to the emergency preparedness department during calendar years 2002 and 2003
- Four quality assurance audits and assessments
- Ten drill and exercise drill reports

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspector used the requirements in 10 CFR Part 20 and the licensee's procedures required by Technical Specifications as criteria for determining compliance. The inspector interviewed licensee personnel and reviewed:

- Current 3-year rolling average collective exposure

- Site-specific trends in collective exposures, plant historical data, and source-term measurements
- Site-specific ALARA procedures
- Six work activities of highest exposure significance completed during the last outage
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Intended versus actual work activity doses and the reasons for any inconsistencies
- Postjob (work activity) reviews
- Assumptions and basis for the current annual collective exposure estimate, the methodology for estimating work activity exposures, the intended dose outcome, and the accuracy of dose rate and man-hour estimates
- Method for adjusting exposure estimates, or replanning work, when unexpected changes in scope or emergent work were encountered
- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding
- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry
- Self-assessments, audits, and special reports related to the ALARA program since the last inspection

Either because the conditions did not exist or an event had not occurred, no opportunities were available to review the following items:

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas
- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

The inspector completed 13 of the required 15 samples and 1 of the optional samples.

b. Findings

1. Introduction. The inspector identified a Green finding because the licensee failed to plan and control dose or provide ALARA Committee oversight for the work activity that accrued the largest portion of the refueling outage dose.

Description. The collective dose accrued during Refueling Outage RE22 was 214 person-rem. Special Work Permit 2005-1072, "Drywell-General Assess/Limited Maintenance," accrued approximately 38 person-rem (37.9649). This was more than any other special work permit or radiological job package used during the 2005 refueling outage. An assortment of tasks were conducted using this special work permit, and workers were allowed to receive up to 300 millirems per individual entry while working in accordance with this special work permit. No limit was placed on the total dose this special work permit could accrue.

Under the licensee's process, the development of a "radiological job package" was necessary before a dose estimate was formulated and a job plan was initiated. Additionally, only radiological job packages were reviewed by the ALARA Committee. Guidance for the development of radiological job packages came from Procedure 9.ALARA.5, "ALARA Planning and Controls," Revision 13. Section 3.4 of this procedure stated that a radiological job package and a specific radiation work permit *"should"* be developed if the expected accumulated exposure for the work activity is greater than or equal to 0.5 person-rem or if the work activity will occur in areas with dose rates greater than or equal to 1 rem per hour at 12 inches. The licensee established no expectation of ALARA performance for Special Work Permit 2005-1072, in the form of a projected dose, but during previous refueling outages in 2001 and 2003, the analogous special work permits accrued 15.3 person-rem and 8.4 person-rem, respectively. Additionally, Special Work Permit 2005-1072 allowed work in areas with maximum dose rates of 1.8 rem per hour. Despite the historical dose information and the potential radiological areas that could be entered, the licensee did not develop a radiological job package for this special work permit. When the inspector asked why a radiological job package was not developed for Special Work Permit 2005-1072, the ALARA supervisor referred to a note in the procedure which stated, "Specific RWPs do not necessarily require a [radiological] job package be developed." The inspector concluded that the permissive language in the procedure contributed to the licensee's failure to establish a job plan and dose estimate or to provide review and oversight by the ALARA Committee.

Analysis. The failure to produce a job plan and dose estimate, limit the ultimate dose, and provide ALARA Committee review and oversight for Special Work Permit 2005-1072 is a performance deficiency.

This special work permit accrued a dose of approximately 38 person-rem. According to Manual Chapter 0308, Appendix C, "The 5 person-rem criterion represents a level of actual dose associated with a work activity at which it is reasonably expected that the licensee will, at a minimum, apply measures to review and plan work, track dose and, if practical, to reduce exposures . . . The 25 person-rem criterion in the significance determination process represents a level of actual dose associated with a work activity at which it is reasonably expected that there will be review and oversight by licensee management to confirm the adequacy of ALARA measures that are being applied."

Manual Chapter 0308, Appendix C, also states, "A mismatch between the planned, intended dose and the actual dose experienced in completing a work activity is an indication of a possible program weakness or failure . . . Planned or intended collective dose can be the results of a realistic dose estimate (or projection) established during ALARA planning or the dose expected by the licensee (i.e., historically achievable) for the reasonable exposure control measures specified in ALARA procedures/planning." Since the licensee had no expectation of the potential dose, the inspector compared the actual dose with historical doses and found that the 2005 doses exceeded the historical totals by more than 50 percent.

This finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute (ALARA planning/estimated dose) and affected the associated cornerstone objective in that the failure to plan and control radiation dose affected the licensee's ability to ensure adequate protection of worker health and safety. The finding involved a failure to maintain or implement, to the extent practical, procedures or engineering controls needed to achieve occupational doses that were ALARA and that resulted in unplanned, unintended occupational collective dose for a work activity.

When processed through the Occupational Radiation Safety Significance Determination Process, this ALARA finding was found to have no more than very low safety significance because the finding was related to ALARA, but the licensee's 3-year rolling average collective dose was not greater than 240 person-rem.

Enforcement. No violation of regulatory requirements occurred. The finding was documented in the licensee's corrective action program as CR-CNS-2005-2985. This finding is identified as Finding (FIN) 05000298/2005003-03, Failure to plan and control dose or provide ALARA Committee oversight for Radiation Work Permit 2005-1072.

2. Introduction. The inspector identified a Green ALARA finding because inadequate planning resulted in the collective dose of a work activity that exceeded 5 person-rem and exceeded the dose estimation by more than 50 percent.

Description. The licensee developed Radiological Job Package 2005AL-03 for work related to removal of sludge from the torus. The radiological job plan (Form CNS RP-49) called for hanging the spent torus desludge filters from the catwalk handrail to drain the water, measuring the dose rate on each filter, lifting the filters from inside the torus, and transferring them to a high integrity container for shipment. However, as the filters were being transferred to the high integrity container, the licensee found that there was insufficient capacity in the container for all the filters. According to the postjob review, after 20 filters were loaded, it became obvious that all 43 filters would not fit into the container. The loading personnel made an attempt to adjust the filters. This resulted in extra dose and the loading of only three additional filters into the container. Because not all of the filters could be removed at that time, the remaining filters had to be placed back into the water and the process had to be repeated when another high integrity container was obtained. This resulted in more collective dose than planned. Instead of 3.2 person-rem, the work activity accrued nearly 5.7 person-rem. The licensee's postjob review identified the miscalculation of the capacity of the vendor-supplied high integrity container as the primary cause for the dose exceeding the projected value.

Analysis. The failure to properly plan the work activity and verify information such as the capacity of a shipping container is a performance deficiency. This finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute (ALARA planning/estimated dose) and affected the associated cornerstone objective in that the failure to control collective dose affected the licensee's ability to ensure adequate protection of worker health and safety from exposure to radiation. The finding involved a failure to maintain or implement, to the extent practical, procedures or engineering controls needed to achieve occupational doses that were ALARA and that resulted in unplanned, unintended occupational collective dose for a work activity. When processed through the Occupational Radiation Safety Significance Determination Process, this ALARA finding was of very low safety significance because the finding was related to ALARA, but the licensee's 3-year rolling average collective dose was not greater than 240 person-rem.)

Enforcement. No violation of regulatory requirements occurred. The finding was documented in the licensee's corrective action program as CR-CNS-2005-2969. This finding is identified as FIN 05000298/2005003-04, Failure to maintain collective doses associated with Radiological Job Package 2005AL-03 ALARA.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspector sampled licensee submittals for the performance indicators listed below for the period of July 1, 2004, through March 31, 2005. The definitions and guidance of Nuclear Engineering Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the licensee's basis for reporting each data element in order to verify the accuracy of performance indicator data reported during the assessment period. Licensee performance indicator data were also reviewed against the requirements of Procedure 0-PI-01, "Performance Indicator Program," Revision 16, and Emergency Preparedness Desk Guide 2, Attachment G-1, "Performance Indicator Guide," Revision 11.

Emergency Preparedness Cornerstone:

- Drill and Exercise Performance
- Emergency Response Organization Participation
- Alert and Notification System Reliability

The inspector reviewed a 100 percent sample of drill and exercise scenarios and licensed operator simulator training sessions, notification forms, and attendance and critique records associated with training sessions, drills, and exercises conducted during the verification period. The inspector reviewed selected emergency responder qualification, training, and drill participation records. The inspector reviewed alert and notification system testing procedures and a 100 percent sample of siren test records. The inspector also interviewed licensee personnel responsible for collecting and evaluating performance indicator data.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Resident Inspector Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed a selection of condition reports written during the inspection period to verify the licensee was entering conditions adverse to quality into the corrective action program at an appropriate threshold. Additionally, the inspectors verified that condition reports were appropriately categorized and dispositioned in accordance with the licensee's procedures and, in the case of significant conditions adverse to quality, to review the adequacy of licensee root cause determinations, extent of condition reviews, and implemented corrective actions. The following condition report was reviewed in depth during this period (one sample):

- CR-CNS-2005-02732, apparent cause regarding failure of SW Booster Pump A dated May 12.

b. Findings

Introduction. A Green NCV was identified regarding the failure to take adequate corrective actions for degraded conditions on the SW booster pump system.

Description. On April 5, the SW Booster Pump (SWBP) A outboard bearing oil level was found to be above the maximum on the oil level sight glass. After investigating, it was determined that the increase in oil level was the result of water intrusion into the outboard bearing housing. The water from a leaking mechanical seal had accumulated in the outboard seal catch basin due to plugging of the basin drain line, which resulted in water filling the catch basin and passing though the outboard bearing housing into the bearing. SWBP A was declared inoperable as a result of the water intrusion. Immediate corrective actions included replacing the bearing oil and cleaning of all SWBP seal catch basin drains. The licensee conducted an apparent cause determination, which discovered that water leaking from the mechanical seal caused a precipitate buildup in the catch basin drain line. The licensee concluded that the buildup in the drain line was a result of an inadequate preventive maintenance (PM) frequency to clean the drains and the amount of leakage from the mechanical seal.

A similar event occurred in February 2004 when SWBP D was discovered with water intrusion into the bearing housing. The licensee's response to that event was similar; the SWBP D basin drain line was cleaned and the bearing oil was replaced. An apparent cause determination was performed which concluded that the debris buildup occurred over a long period of time and could be managed by the PM program. The PM to clean and inspect the basins was increased from a condition based frequency to a 36-week frequency. Based on the past condition of the basin drains, the licensee concluded that the 36-week PM frequency was adequate to prevent clogging. However, this corrective action only addressed the symptoms of the adverse condition (buildup of debris in the drain line), not the root cause (the fact that the mechanical seal was leaking). At the time of the clogging in April 2005, it had only been 33 weeks since the basin drain had been cleaned. Following the April clogging, the licensee increased the PM frequency from a 36-week to 4-week frequency. Notification 10325620 was written in July 2004 to document leakage from the SWBP A mechanical seal; however, corrective actions for this condition are not scheduled for completion until December 2005.

Analysis. The inspectors concluded that the water intrusion into the SWBP A bearing, which rendered the pump inoperable, was a significant condition adverse to quality and was considered more than minor since it affected the Mitigating Systems Cornerstone attribute of equipment operability, availability, and reliability. Based on the Significance Determination Process Phase 1 screening, this finding was determined to have very low safety significance, since it did not represent the actual loss of a safety function.

This finding had crosscutting aspects associated with problem identification and resolution. This assessment was based on the fact that previous corrective actions only addressed the symptoms of the adverse condition, not the root cause.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition. Failure of the SWBP A was considered a significant condition adverse to quality since it adversely impacted the ability of a risk significant safety system to perform as designed. Corrective actions for SWBP D failure in February 2004 failed to preclude an additional failure in April 2005. Because this violation was of very low safety significance and was entered in the corrective action program as CR-CNS-2005-02732, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000298/2005003-05, Failure to take Adequate Corrective Actions for degraded conditions on Service Water Booster Pump System.

.2 Semiannual Trend Review

a. Inspection Scope

The inspectors performed a semiannual assessment of trends in the licensee's corrective action program to determine if any more significant safety issues existed. Specifically, the inspectors reviewed the licensee's corrective action program database to determine if the licensee had identified trends in any of the following areas:

- Equipment fires
- Configuration control errors
- 4160 V breakers
- Foreign material exclusion control issues
- Age related failures of electronic subcomponents
- Grounds on 480 V motor control centers (MCC)

These areas were chosen based on information gathered by the inspectors during daily plant status reviews over the previous 6 months. For those areas where trends were documented in the corrective action program, the inspectors verified that the licensee had corrective actions planned or in place to address the trend. For the remainder of the issues in the scope of this inspection, the inspectors reviewed control room logs, system health reports, quality assurance audits, and department self-assessments and interviewed selected licensee staff to determine if any adverse trends existed.

b. Findings and Observations

The inspectors concluded that, in general, the licensee had adequately identified trends in the areas within the scope of this inspection; however, these trends were not always entered into the corrective action program or formally evaluated to determine the need for corrective actions. For example, the inspectors searched the corrective action database for the term "fire" over the last 12 months. This search yielded results indicating that there had been six fires at Cooper Nuclear Station in the first 5 months of 2005. After discussions with the licensee regarding a possible adverse trend in this area, the licensee determined that there had been 25 fire events at the plant in the past 5 years. Furthermore, there was no documentation of this potential adverse trend in the corrective actions program nor had a formal common cause analysis ever been performed to identify any programmatic contributors to the large number of fires at the station. As a result of these discussions, the licensee documented a potential trend in the corrective action program as CR-CNS-2005-03906 and initiated a common cause analysis.

An additional example involved the large number of intermittent grounds on MCCs in the plant. The inspectors searched control room logs and the corrective action program for the term “ground” over the past 12 months. This search yielded 21 grounds, 16 of which occurred in the first 5 months of 2005. Several of the grounds were only documented in the control room logs with no corresponding condition reports. The most prevalent grounds occurred on MCCs A and B with a total of 11 grounds recorded in 2005. The licensee did document an adverse trend in the number of grounds on 125 Vdc Battery Charger A in CR-CNS-2005-1407; however, no other documentation of a potential adverse trend existed in the corrective action program. As a result of discussions regarding a possible adverse trend, the licensee documented the trend in the corrective action program as CR-CNS-2005-03909 and initiated a common cause analysis.

.3 Emergency Preparedness Annual Sample Review

a. Inspection Scope

The inspector selected 25 notifications and condition reports for detailed review. The reports were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspector evaluated the condition reports against the requirements of Procedures 0.5 CR, “Condition Report Initiation, Review and Classification,” Revision 1, and Procedure 0.5NAIT, “Corrective Action Implementation and Nuclear Action Item Tracking,” Revision 23.

b. Findings and Observations

No findings of significance were identified.

.4 Identification and Resolution of Problems Crosscutting Aspects of Findings

Sections 1R14 and 4OA2.1 describes a finding with crosscutting aspects associated with problem identification and resolution.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report 05000298/2005001-00, Reactor Scram Due to Reactor Level Transient and Inadvertent Rendering of High Pressure Coolant Injection Inoperable

On April 15, 2005, the reactor automatically scrammed on low reactor vessel water level due to the failure of the master level controller. No performance deficiencies were identified with regard to the cause of the scram; however, Section 1R14 discusses a noncited violation with respect to operator performance during scram recovery actions. This licensee event report is closed.

4OA4 Crosscutting Aspects of Findings

Sections 1R14 and 4OA5.3 describes a finding with human performance aspects.

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000298/2004015-01: Simulator Fidelity.

In order to disposition Unresolved Item 05000298/2004015-01, nine major fidelity issues listed as open in the simulator discrepancy reports and documented in the associated report as part of the unresolved item had to be tested. Of these nine items, only two were substantiated: (1) Item 3, which stated that "Use of boron in emergency operating procedures associated with Anticipated Transient without Scram scenarios produces an immediate homogeneous reduction in steady state reactor power on the simulator"; and (2) Item 8, which stated that "[r]eactor water level indicators do not become erratic under rapid depressurization conditions. Erratic level indications should occur because the rapid depressurization results in reference leg flashing." The inspectors reviewed the original acceptance test procedures and documentation and concluded that the simulator was not designed for either of these features and, as such, these items should have been placed in the enhancements report, not the discrepancies report provided to the inspectors in November of 2004. Therefore, Unresolved Item 05000298/2004015-01 is closed. The issues discussed with the licensee related to the discrepancy characterization and disposition that resulted in this followup inspection are being corrected under Corrective Action Program Number LO-CNSLO-2005-00010 CA-0005.

.2 (Closed) Unresolved Item 05000298/2004015-02: Acceptance Criteria for Simulator Performance Testing.

In order to disposition Unresolved Item 05000298/2004015-02, "Acceptance Criteria for Simulator Performance Testing," the inspectors reviewed 11 simulator performance test records, which are identified in this report (5 transient tests, 5 malfunction tests, and 1 steady state test). Two of these were originally reviewed during the 2004 inspection (T4 and T10) while the remaining test records (T5,T6,T8, MS03A, ATWS, FW-14, FW-15, and SDP-05-0039) were sampled from the closed discrepancies documented between November 2004 and April 2005. The steady state test identified was a shutdown cooling malfunction that, because of a discrepancy, caused a continuous cooldown of the plant with the residual heat removal heat exchanger outlet valves closed. The work package for the repair and test package was also SDP-05-0039. The final test record (and the largest closed risk-significant discrepancy) reviewed was the failure of the output breaker for Emergency Diesel Generator 2 to close on the bus in certain electric plant failures. This test package (SWP-04-0028) was also run as a scenario (S-1) in combination with a loss-of-offsite power in the simulator by the inspectors. No issues were found for any of the identified records.

.3 (Closed) Unresolved Item 05000298/2004005-02: Review Safety Significance of Degraded Startup Flow Control Valves

Introduction. A Green self-revealing finding was identified involving the failure to perform an adequate design change for the reactor feed system startup flow control valves.

Description. In September and October of 2004, several reactor feed pump trouble alarms, related to startup flow control valve abnormal operation, were received in the control room. These valves are used to control feedwater flow to the reactor vessel during low power operation. Troubleshooting activities determined that the alarms were caused by failures of the valve actuators. Both valve actuators were replaced on September 25, 2004; however, during postmaintenance testing, the valves did not cycle as expected. Additional troubleshooting determined that the valve piston O-ring seals were degraded, resulting in erratic operation of the valves.

The licensee’s apparent cause investigation discovered that the actuator O-rings were square in shape, indicating that their temperature rating had been exceeded. These O-rings had a temperature rating of 175EF. During normal operation, the nominal reactor feedwater temperatures is 360EF, which exceeded this rating. These O-rings were installed in October of 2002 under modification CED 6008140, which did not adequately consider the type of material that would be compatible with normal operating temperatures.

Analysis. The inadequate design review of the reactor feed pump startup flow control valve modification (CED 6008140) was considered to be a performance deficiency. This finding affected the Mitigating Systems and Initiating Events Cornerstones and is more than minor because it affected the attribute of design control. Based on the results of a Significance determination Process Phase 1 evaluation, the finding affected two cornerstones and screened to a Phase 2 evaluation. The inspectors requested assistance from a senior reactor analyst for the Phase 2 evaluation. The senior reactor analyst determined that the evaluation would require a departure from the Phase 2 process. Therefore, this evaluation constituted a Phase 3 analysis. The analyst made the following assumptions:

- The startup flow control valves (FCVs) are normally used only during reduced-power operations during plant startups and shutdowns. If an initiating event were to occur at these conditions, decay heat load would be lower than if an event occurred at full power. Therefore, use of the at-power Significance Determination Process Notebook worksheets would provide a bounding result with respect to mitigating systems success criteria.

- The finding affected the feed portion of the power conversion system (PCS) feed. The startup FCVs are in the feedwater flowpath. When the valves were in service during startup or shutdown conditions, unexpected opening or closure of the valves could cause a loss of feedwater control, causing a plant transient. Therefore, the frequency of a transient increased as a result of the finding.

- The condition existed for greater than one year. However, the actual exposure time window during which the performance deficiency increased the initiating event likelihood was limited to when the startup FCVs were in service. The analyst assumed that the startup FCVs would reasonably be expected to be in service approximately 24 hours per year. This placed the exposure time window in the <3 day category.

- Table 1 of the Risk-Informed Inspection Notebook identified that the initiating event likelihood (IEL) for a transient with an exposure time window of <3 days was 3. However, because the finding increased the likelihood of an initiating event, the analyst applied Usage Rule 1.2 and increased the IEL by one order of magnitude, to 2.

- Following a transient, in accordance with system design and documented procedure, operators would be able to open the main feedwater valves after a 3-minute time delay and resume reactor water level control using the main feedwater pumps. Therefore, the analyst assumed that full mitigation credit remained for the PCS safety function in the Risk-Informed Inspection Notebook transient worksheet (and remained operator-action limited with credit = 3).

- The likelihood of other initiating events (for example, loss-of-coolant accidents) occurring simultaneously with failure of the startup FCVs was considered too remote during the 24-hour exposure window to warrant analysis. Therefore, only the transient worksheet was evaluated.

Based on the above assumptions, each sequence in the Risk-Informed Inspection Notebook Transient worksheet was evaluated. The results are summarized below.

TRANSIENT WORKSHEET RESULTS			
Sequence	IEL	Mitigating Functions	Result
1	2	PCS-CHR-LI	11
2	2	PCS-CHR-CV	10
3	2	PCS-HPI-LPI	14
4	2	PCS-HPI-DEP	10

By application of the Counting Rule, the internal event risk contribution of this finding was less than 1×10^{-7}.

The plant-specific Significance Determination Process worksheets do not currently include initiating events related to fire, flooding, severe weather, seismic, or other external initiating events. In accordance with Manual Chapter 0609, Appendix A, Attachment 1, step 2.5, "Screening for the Potential Risk Contribution Due to External Initiating Events," experience with using the site-specific Risk-Informed Inspection Notebooks has indicated that accounting for external initiators could result in increasing the risk significance attributed to an inspection finding by as much as one order of magnitude. The analyst determined that an evaluation of external risk would not be required because the result of the Phase 3 analysis indicated that the risk was less than 1×10^{-7}. Therefore, an increase in the risk by an order of magnitude would not result in the significance of the finding crossing the 1×10^{-6} threshold.

In accordance with Manual Chapter 0609, Appendix A, Attachment 1, step 2.6, "Screening for the Potential Risk Contribution Due to LERF," the analyst determined that the finding was not significant from a large early release frequency perspective and no further evaluation was necessary because the Phase 3 result provided a risk significance estimation of less than 1×10^{-7}.

The performance deficiency resulted in a finding that was of very low risk significance (Green).

This finding had crosscutting aspects associated with human performance. This assessment was based on the fact that engineering procedures reflected current guidance in evaluating system environmental conditions related to installing modifications and that appropriate training had been conducted, yet personnel still failed to follow the procedure.

Enforcement. No violation of regulatory requirements occurred because the reactor feed pump startup flow control valves are not classified as safety-related. The licensee entered this finding into their corrective action program as CR-CNS-2004-06997. This finding is identified as FIN 05000298/2005003-06, Inadequate Design Review of System Modification.

.4 Temporary Instruction 2515/163, Operational Readiness of Offsite Power

The inspectors assessed the operational readiness of the offsite power system serving Cooper Nuclear Station in accordance with Temporary Instruction 2515/163. The results of this inspection were communicated to the NRC's Office of Nuclear Reactor Regulation on June 1, 2005. No findings of significance were identified.

4OA6 Meetings, Including Exit

On April 15, 2005, the health physicist inspector presented the inspection results to Mr. S. Minahan, General Manager of Plant Operations, and other members of his staff who acknowledged the findings. On May 6, 2005, the inspectors conducted a telephonic exit on the final inspection results with Mr. J. Roberts, Director, Nuclear Safety Assurance, and other members of the licensee's staff.

On April 22, 2005, the operations inspectors debriefed the inspection results to Mr. S. Minahan, General Manager of Plant Operations, and other members of licensee management. On May 19, 2005, the inspectors conducted a telephonic exit of the inspection results to Jerry Roberts, Director of Nuclear Safety Assurance who acknowledged the findings.

On May 18, 2005, the emergency preparedness inspector presented the inspection results to Mr. J. Roberts, Director, Nuclear Safety Assurance, and other members of his staff who acknowledged the findings.

On July 7, 2005, the resident inspectors presented the results of the inspection activities to Mr. S. Minahan and other members of his staff who acknowledged the findings.

The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Bednar, Emergency Preparedness Manager
C. Blair, Engineer, Licensing
D. Cook, Technical Assistant to General Manager
S. Minahan, General Manager of Plant Operations
T. Chard, Radiological Manager
K. Chambliss, Operations Manager
J. Christensen, General Manager of Support
R. Estrada, Corrective Actions Manager
J. Flaherty, Site Regulatory Liaison
P. Fleming, Licensing Manager
W. Macecevic, Work Control Manager
J. Roberts, Director, Nuclear Safety Assurance
R. Shaw, Shift Manager
J. Sumpter, Senior Staff Engineer, Licensing
K. Tanner, Shift Supervisor, Radiation Protection
R. Hayden, Emergency Preparedness Staff
T. Chard, Manager, Radiation Protection
R. Edington, Vice President
S. Blake, Manager, Quality Assurance
K. Fili, Manager, Nuclear Projects
D. Kimbell, Outage Manager
G. Kline, Director, Engineering

NRC Personnel

L. Ricketson, Senior Health Physicist
S. Schwind, Senior Resident Inspector
S. Cochrum, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000298/2005003-01	NCV	Reactor Operation in Excess of Licensed Thermal Power Limits (Section 1R14)
05000298/2005003-02	NCV	Inadequate Corrective Actions Result in High Pressure Coolant Injection System being Rendered Inoperable (Section 1R14)

05000298/200503-03	FIN	Failure to plan and control dose or provide ALARA Committee oversight for Radiation Work Permit 2005-1072 (Section 2OS2)
05000298/200503-04	FIN	Failure to maintain collective doses associated with Radiological Job Package 2005AL-03 ALARA (Section 2OS2)
05000298/2005003-05	NCV	Failure to take Adequate Corrective Actions for degraded conditions on Service Water Booster Pump System (Section 4OA2)
05000298/2005003-06	FIN	Inadequate Design Review of System Modification (Section 4OA5)

Closed

05000298/2004015-01	URI	Simulator Fidelity (Section 4OA5)
05000298/2004015-02	URI	Acceptance Criteria for Simulator Performance Testing (Section 4OA5)
05000298/2004005-02	URI	Inadequate Design Review of System Modification (Section 4OA5)
TI 2515/163	TI	Operational Readiness of Offsite Power (Section 4OA5)
05000298/2005001-00	LER	Reactor Scram Due to Reactor Level Transient and Inadvertent Rendering of High Pressure Coolant Injection Inoperable (Sections 1R14 and 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 2OS2

Corrective Action Documents (Condition Reports)

2005-00982, 2005-01321, 2005-01420, 2005-02335, 2005-02459

Audits and Self-Assessments

LO-CNSLO-2005-00029, ALARA

Radiation Job Package

2005AL-03 Torus Desludge/Inspect/Repair
2005AL-04 Feedwater Checkvalve Replacement
2005AL-06 Refuel Floor Activities
2005AL-09 ISI/EC Activities for RE-22

2005AL-11 Drywell Shielding
2005AL-12 –4 A, B, C, & D Insulation Installation

Procedures

0.ALARA.1 CNS ALARA Program, Revision 3
0.ALARA.2 ALARA Organization and Management, Revision 7
9.ALARA.4 Radiation Work Permits, Revision 5
9.ALARA.5 ALARA Planning and Controls, Revision 13
9.ALARA.12 Hot Spot Reduction Program, Revision 1

ALARA Committee Minutes

Meeting 2004-08 - June 4, 2004
Meeting 2004-09 - June 28, 2004
Meeting 2004-10 - July 13, 2004
Meeting 2004-11 - August 23, 2004
Meeting 2004-12 - November 9, 2004
Meeting 2004-13 - November 9 [sic], 2004

Section 1EP2

Alert and Notification System Design Report, Revision 12

Section 1EP3

Evaluations for Call-In Drills conducted:
February 16, 2004
April 13, 2004
June 1, 2004
August 19, 2004
October 7, 2004
December 13, 2004
February 24, 2005
April 12, 2005

Section 1EP5

Procedure 0.5TREND, “Corrective Action Program Trending,” Revision 6
CAP Desk Guide 11, “CNS Risk Significance Determination Screening Process,” Revision 5
QA Audit 04-04, Emergency Preparedness, April 8, 2004
QA Audit 05-03, Emergency Preparedness, April 19, 2005
CNS Quality Assurance Surveillance Report S403-0402
Dose Assessment Review, April 27, 2004
Notifications: 10258288, 10267318, 10267630, 10272637, 10323621, 10325028, 10326429, 10326517, 10328422, 10332946, and 10350609

Condition Reports: 2004-6185, 2004-6189, 2004-6889, and 2005-02549

Apparent Cause Analysis RCR 2003-2029

Evaluation Reports for Exercises and Drills conducted:
August 28, 2003
September 24, 2003
October 8, 2003
December 17, 2003
June 30, 2004
August 5, 2004
August 24, 2004
September 21, 2004
October 26, 2004
December 13, 2004

2004 & 2005 Emergency Preparedness Drill Schedules
Procedure 5.7.1, "Emergency Classifications," Revision 31
Procedure 5.7.2, "Emergency Direction EPIP," Revision 21
Procedure 5.7.6, "Notification," Revision 40
Procedure 5.7.20, "Protective Action Recommendations," Revision 17

Section 1R11

Station Procedures Governing Simulator Discrepancy Process

CNS NTP 7.2, "Simulator Configuration Management," Revisions 3 (February 2004) and 4 (November 2004)

CNS NTP 7.3, "Simulator Physical Fidelity," Revisions 3 (July 2003) and 4 (December 2004)

CNS NTP 7.4, "Simulator Performance Test Documentation," Revision 2 (December 2004)

CNS NTP 7.5, "Simulator Performance Review Committee," Revisions 2 (March 2004) and 3 (February 2005)

CNS NTP "Desk Guide 4.1 For Simulator Performance Testing," Revision 2 (November 2002)

CNS NTP "Desk Guide 4.1 For Simulator Performance Testing," Revision 3 (December 2004)

CNS NTP "Desk Guide 1.1 For Simulator Work Packages," Revision 3 (January 2004)

CNS NTP "Desk Guide 1.5 For Processing Simulator Work Requests," Revision 4 (January 2005)

CNS NTP "Desk Guide 2.1 For Simulator Software Controls," Revision 5 (May 2002)

Simulator Open Discrepancy Report (ALL), November 2004

Simulator Open Discrepancy Report (DR), March 18, 2005

Simulator Closed Discrepancy Report (DR) for period August 2004 - March 2005, generated March 2005

Simulator Plant Mods Implemented on the Simulator Report for November 2004-March 2005, generated December 2004

Simulator Plant Mods Implemented on the Simulator Report for November 2004-March 2005, generated December 2004

Internal Memoranda Regarding Simulator Performance and Updates

“Simulator Internal Self-Assessment Report SA03001," dated March 2003
“Management Overview of Simulator Annual Performance Tests 2004,” December 6, 2004
“IP 711111.11C Nov 2004 Inspection-Response from Simulator Group,” November 24, 2004
“Simulator Performance Review Committee Minutes,” February 24, 2003
“Simulator Performance Review Committee Minutes,” November 3, 2003
“Simulator Performance Review Committee Minutes,” November 13, 2003
“Simulator Performance Review Committee Minutes,” March 4, 2004
“Simulator Performance Review Committee Minutes,” April 28, 2004
“Simulator Performance Review Committee Minutes,” June 28, 2004
“Simulator Performance Review Committee Minutes,” October 15, 2004
“Simulator Configuration Update-Production Training Load 0402,” February 20, 2004
“Simulator Configuration Update-Production Training Load 0403,” March 29, 2004
“Simulator Configuration Update-Production Training Load 0404,” May 10, 2004
“Simulator Configuration Update-Production Training Load 0405,” May 24, 2004
“Simulator Configuration Update-Production Training Load 0406,” August 3, 2004
“Simulator Configuration Update-Production Training Load 0407,” August 20, 2004
“Simulator Configuration Update-Production Training Load 0408,” October 7, 2004
“Simulator Configuration Update-Production Training Load 0409,” October 21, 2004

Other Procedures

CNS Startup Procedure 2.1.1, Revision 117
CNS Administrative Procedure 0-EBS-NOT for “SAP Notifications,” Revision 18 (January 2005)

Dynamic Scenario Tests

INT008-03-14
COR008-43-11/1548
SKL051-51-50
SKL012-42-18
SKL012-47-19
SKL012-42-11
SKL051-51-100
SKL051-51-102
INT008-03-34
COR008-43-10/1547
Licensed Operator Requalification Evaluated Scenario 1
Licensed Operator Requalification Evaluated Scenario 2

Nonlibrary Scenario Tests

S-1, Loss of offsite Power, delayed to a Station Black-Out

S-2, LOCA, modified for largest DW temperature

1. Ensure DW temperature exceeds 280EF for this event
2. Ensure Max DW temp reached consistent with FSAR (304EF).

S-3, ATWS, modified for injection to monitor power/pressure

1. Ensure Reactor Power goes down, APRM's downscale
2. Ensure Reactor Pressure continues to go down beyond 600psig with six open safety relief valves and injection stopped

S-4, DBA LOCA, modified to monitor Reactor Water level recovery

1. DBA LOCA with only one Core Spray Pump running
2. Ensure recovery of Reactor Water Level to 2/3 core height
3. Ensure one Core Spray pump can not continue to raise level beyond TAF

S-5, Scram Discharge Instrument Volume (SDIV) Rupture

1. Test plant response
2. Validate proper secondary containment response (radiation, temperature, sump levels all increase)

S-6, ATWS, modified for 5% power initiation

1. Monitor Reactor water level for variations
2. Shift to EOP-7A for power/level control

S-7, ATWS Test with malfunctions listed numerically below:

1. Fail to scram (hydraulic lock - SDV solid)/ARI
2. Gp 1 isolation over-ride
3. Feedwater Isolation over-ride
4. Inhibit ADS
5. HPCI in over-ride/fail to start
5. End of Cycle Recirc Pump Trip Over-ride
6. Attempt to drive in a control rod
7. Attempt to withdraw a control rod
8. Runback the recirc pumps to minimum
9. Trip one recirc pump and observe jet pump flows
10. Trip second recirc pump
11. Trip one feed pump
12. Place the FWLCS in manual
13. Reduce RPV level to TAF or < 3% on APRMs
14. Raise level to where the FWLCS can be put in AUTO - observe shrink and swell

15. Place FWLCS in AUTO
16. Initiate SLC - observe power change
17. Close the MSIVs

S-8 Custom DR malfunction set to test significant DR's from DR report identified:

1. Trip condensate pumps until only 2 are running
2. Restart condensate pump or reset
3. Trip one feedpump
4. Restart feedpump or reset
5. With the Droop switch in ISOCH EDG-2, open supply breaker to bus 1G
6. Reclose supply breaker to 1G or reset
7. Take SJAEs out of service (isolate) and observe vacuum
8. Start HPCI Aux Oil Pump, Open MO-19, let ramp generator time out, open MO-14.
9. Close HPCI Supply valves and monitor pressure decay
10. Isolate extraction steam to Heater 5A
11. Trip condenser circ water pumps one at a time

S9- Group 1 Isolation Test

1. Initiate a Group 1 isolation
2. Observe MSIVs, FWIVs, and RPV level.

S10- Design Basis LOCA

1. Over-ride HPCI, RCIC, LPCS, and RHR. Keep CRD running

2. Initiate recirc line break DBA

3. Put in freeze when primary containment pressure stabilizes. Record primary containment pressures and temperatures. Record RPV level

4. Compare to FSAR design bases analysis for DBA LOCA

S11- Turbine Trip from 20% Power

1. Drive an out-of-sequence rod
2. Place the RWM in bypass
3. Trip the main turbine

<u>Transient Tests</u>

T-4, Simultaneous Trip of both Recirc Pumps
T-5, Single Recirc Pump Trip
T-6, Main Turbine Trip
T-8, LOCA (design basis accident, recirc pump suction break)
T-10, Closure of all MSIV's with stuck open safety relief valve

Malfunction Tests

SWP-04-0028, various EG2 failures, including o/p breaker failure to close in ISOL
MS03A, Main Steam Line Rupture
ATWS, Anticipated Transient without Scram
FW-14, Loss of Condensate Pump
FW-15, Loss of 1 Feed Pump

Steady State Tests

Shutdown Cooling Work Package SDP-05-0039-fixed discrepancy on HX outlet valve.

Alternate Training Instruction for Simulator Discrepancies

Simulator Significant Discrepancies with Alternate Training Methods report, April 19, 2005
SKL012-06-01 Lesson plan Deviations for Simulator Discrepancies, Revision 98, dated April 2005

Miscellaneous

Simulator Upgrade Gant Chart, dated November 2004
Simulator Upgrade Gant Chart, dated March 2005
Cooper Public Meeting Presentation with NRC, February 2005

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
CFR	*Code of Federal Regulations*
FCV	flow control valve
FEMA	Federal Emergency Management Agency
FIN	finding
HPCI	high pressure coolant injection
IEL	initiating event likelihood
MCC	motor control center
NCV	noncited violation
NRC	U.S. Nuclear Regulatory Commission
PCS	power conversion system
PM	preventive maintenance
RFP	reactor feed pump
SW	service water
SWBP	service water booster pump