EXHIBIT 3

SAMPLE REACTOR INSPECTION REPORT

The sample inspection report can be found at the following website:

http://nrr10.nrc.gov/rop-digital-city/sampleIR.pdf

The inspection report is a representative sample inspection report and not an all inclusive guide. It is based on a fictional reactor licensee and a fictional integrated inspection period. The report contains realistic findings (generally for a BWR); however, any resemblance to an existing facility or actual events is coincidental. The user should recognize that the sample report was assembled from various facility reports and newly drafted material, so terminology of facility items may inherently be inconsistent when the report is viewed as a whole (i.e. the definition of corrective action documents; procedure numbering schemes; etc.). Some text is underlined when choices need to be made. The sample report also contains text that is in italics and bold which are notes for emphasis to the sample report user and are not to be considered part of the sample report.

This exhibit may be used as a sample report for format and style. It illustrates how to use the standardized inspection report outline, and adheres to the expected internal organization for each report section (as discussed in IMC 0612). Although the sample does not include an example for each baseline inspection program procedure, it does include examples to illustrate the various ways findings would normally be documented.

Pages are numbered continuously through this exhibit. Inspection reports should use separate page numbering for the cover letter, report (beginning with report cover page), and supplemental information.

The font face and size should be Arial 11 for inspection reports.

U.S. NUCLEAR REGULATORY COMMISSION

REGION X

Docket Nos.:	50-998, 50-999
License Nos.:	NPF-01, NPF-02
Report No.:	05000998/2004007 and 05000999/2004007
Licensee:	Greckenshire Power & Light (GP&L)
Facility:	Dirojac Electric Station, Units 1 and 2
Location:	Fridge, North Dakota
Dates:	June 27, 2004 through September 25, 2004
Inspectors:	***Note: Only inspectors who provided an input to the report*** J. Larkin, Senior Resident Inspector J. Henry, Resident Inspector J. Boyle, Senior Health Physicist ***(Note: Optional to use above format to identify specific sections for inspectors other than the residents)***
Approved by:	John J. Miller, Chief Reactor Projects Branch 4 Division of Reactor Projects

(The report, which commences with this page, is an enclosure to the cover letter, and starts as page 1. "Enclosure" should therefore be inserted as a footer at the bottom of each page and flush to the right [not shown].)

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SUMMARY OF FINDINGS

IR 05000998/2004-007, 05000999/2004-007; 06/27/2004 - 09/25/2004; ***(Note: the dates of inspection come after the report #)*** Dirojac Electric Station, Units 1 and 2; Licensed Operator Requalification Program, Maintenance Effectiveness, Operability Evaluations, Permanent Plant Modifications, Surveillance Testing, Access Control to Radiologically Significant Areas, and Event Followup ***(Note: Insp. Procedure or Attachment titles listed only for areas where findings were identified, otherwise just identify the type of inspection e.g., "routine integrated report." Limit is 256 characters - the above exceeds that due to the atypical number of findings in the sample report. Abbreviations can be used as long as they are easy to understand.)***

The report covered a (***use either 13-week or 3-month)*** period of inspection by resident inspectors and an announced inspection by a regional senior health physics inspector. Seven Green findings, all of which were non-cited violations (NCVs), and one AV item with potential safety significance greater than Green, were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. ***(Note: The previous two sentences should be deleted if no findings were identified)*** 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: Initiating Events
(Note: Each finding is self-contained for PIM entry with respect to abbreviations)

- Green. A Green self-revealing non-cited violation (NCV) of Technical Specification 5.4.1.a ***(regulation cited)*** was identified for failure to have an adequate surveillance procedure for calibrating a safety relief valve (SRV) while at power. This resulted in the inadvertent opening of an SRV during the calibration activity. The procedure failed to provide instructions to reset the low-low setpoint logic before applying an input signal to the trip unit. The licensee entered the deficiency with the surveillance procedure into their corrective action (CA) program for resolution. *(First section describes the finding; also include a brief description of the corrective actions taken or planned by the licensee.)*

  (Followed by a brief regulatory and significance evaluation. NOTE present tense of this paragraph) This finding is greater than minor because it had an actual impact of lifting a SRV and therefore could be reasonably viewed as a precursor to a significant event. If the SRV had stuck open, it could have caused a reactor scram. Although the event contributed to the likelihood of a reactor trip, the finding is of very low safety significance because all mitigation systems were available during the use of the surveillance procedure. The cause of the finding is related to the cross-cutting element of problem identification and resolution. (Section 4OA3.3) ***(Note: Briefly describes why greater than minor, provides effect on cornerstone, and states why not greater than green.)***

- Green. A self-revealing NCV was identified for the licensee's failure to comply with 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures and Drawings. As a result of inadequate procedures and poor human performance, a Reactor Building crane trolley was dropped approximately four feet onto the refuel floor while being rigged. The licensee performed a thorough root cause of the event to determine the short and long term corrective actions. There was no permanent structural damage to the refueling floor.

  This finding is greater than minor because it is associated with program and process attributes and affected the objective of the Reactor Safety/Initiating Event Cornerstone to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at power operations. In addition, if left uncorrected, this finding would result in a more significant safety concern. This finding was determined to be a finding of very low safety significance because no initiating event or transient actually occurred, there was no permanent structural damage to the refuel floor, there was no functional degradation, and mitigating capability was not affected. The cause of the finding is related to the cross-cutting element of human performance.(Section 1R17)

Cornerstone: Mitigating Systems

- Green. The inspectors identified an NCV of 10 CFR 50.9 for failure to provide complete and accurate information for one licensed operator on his initial license application. The failure to certify the need for corrective lenses resulted in an incorrect licensing action by the NRC because a license was issued without a restriction to wear corrective lenses. The licensee took prompt corrective action and submitted a letter dated November 19, 2004, requesting lens restriction for the operator's license. The licensee entered this issue into the CA program, and conducted a 100 percent review of all operator medical records to ensure no other discrepancies existed. No other discrepancies were found.

  Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated using the traditional enforcement process. There was no evidence that the operator endangered plant operations as a result of impaired visual acuity while performing licensed duties since the original issuance of his license. However, the regulatory significance was important because the incorrect information was provided under sworn statement to the NRC and impacted a licensing decision for the individual. This issue is documented in the facility licensee's corrective action program as Problem Evaluation Report (PER) 72386. (Section 1R11.1)

- Green. The inspectors identified an NCV of 10 CFR 50.65 (Maintenance Rule) for failing to demonstrate that the performance of the Reactor Motor-

Operated Valve (RMOV) Board 1B was being effectively controlled through the performance of appropriate preventive maintenance such that the system remained capable of performing its intended function. As a result, after it exceeded its Maintenance Rule a(2) performance criteria, the licensee had not established goals nor monitored the performance of the RMOV Board 1B per 10 CFR 50.65a(1). The licensee entered the problem with their failure to monitor the performance of the RMOV into their CA program for resolution.

This finding is more than minor because it affected the reliability objective of the Equipment Performance attribute under the Mitigating Systems Cornerstone. The finding is of very low safety significance because there was no design deficiency, the equipment affected by the board failure either failed in a safe manner or had its redundant equipment functional. (Section 1R12)

- Green. The inspectors identified an NCV for a failure to comply with Technical Specification 3.3.1. when a Loop Control Processor (LCP) failed in Unit 2. The processor failure caused one channel of the reactor protection system to be inoperable and that required the channel to be placed in trip within 6 hours. Because of a licensee position that the processor failure placed all channel bistables in the correct position, operators took no action to trip the channel until approximately 9½ hours after the failure, when preparing to replace the failed processor. The licensee entered their failure to comply with TS into their CA program for resolution.

  This finding was more than minor because it affected the configuration control attribute of the Mitigating Systems Cornerstone in that it reduced the reliability of the required number of operable channels required by the reactor protection system. Had actual plant conditions called for a trip, not taking deliberate operator action to place the inoperable channels in a tripped condition would reduce the likelihood of proper coincident protection system actuation. This finding is of very low safety significance because there was no loss of safety function and the bistables were actually in the tripped condition. (Section 1R15)

- Green. The inspectors identified a NCV of Technical Specification (TS) 5.7.1, which requires that written procedures be implemented covering the activities in the applicable procedures recommended by Regulatory Guide 1.33, including procedures for surveillances. The surveillance procedure for remote shutdown system instrumentation was inadequate because it failed to give guidance for determining instrument operability when an instrument was at the top of scale and at the maximum allowed channel deviation. The performance deficiency resulted in an unexpected TS Limiting Condition for Operation entry. The licensee entered this performance deficiency into their CA program for resolution.

  This finding is greater than minor because it affects the ability of the licensee to monitor the status of the reactor following a control room evacuation and is associated with the Mitigating Systems Cornerstone and the respective attribute of procedure quality. This finding is of very low safety significance because it did not result in a loss of function per Generic Letter 91-18, did not represent an actual loss of safety function, and is not potentially risk-significant due to external events. A contributing cause of the finding is related to the cross-cutting element of human performance. (Section 1R22.1)

- TBD. The inspectors identified an AV for failure to promptly identify and correct binding problems with the Siemens breaker mechanism operated cell (MOC) slide assembly that resulted in the failure of Residual Heat Removal Pump 1A to start on demand. This has potential safety significance greater than very low safety significance and will remain unresolved pending completion of the significance determination process.

  This finding was considered more than minor because, given that Siemens breakers were used in both trains of several emergency core cooling subsystems, the failure to identify and correct a problem that resulted in a pump failure to start on demand could reasonably be viewed as a precursor to a significant event. This finding was also determined to potentially have greater significance because the loss of one train of residual heat removal would result in reduced sump re-circulation capability following a small or medium break size loss-of-coolant accident and no re-circulation capability following the loss of 125-volt DC Battery Board 2. The cause of the finding is related to the cross-cutting element of problem identification and resolution. (Section 4OA2.3)

Cornerstone: Occupational Radiation Safety

- Green. A self-revealing NCV of Technical Specification 5.7.3 was identified because the licensee failed to control a high radiation area with dose rates greater than 1,000 millirems per hour. On July 31, 2004, three workers' electronic alarming dosimeters unexpectedly alarmed when they were exposed to unanticipated radiation levels of approximately 1,700 millirems per hour. The area was not barricaded, conspicuously posted, and did not have a flashing light activated as a warning device. The licensee determined that the three workers received 84, 85, and 95 millirems, respectively. This finding was entered into the licensee's corrective action program.

  This finding is more than minor because it is associated with the Occupational Radiation Safety attribute of exposure control and affected the cornerstone objective, in that not controlling locked high radiation areas could increase personal exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined that the finding was of very low safety significance (Green) because it did not involve: (1) as low as is reasonably achievable planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose (Section 2OS1).

B. Licensee-Identified Violations. ***(Note: The paragraph below is standard language for when licensee-identified violations are documented)***

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

(Note: If no licensee-identified violations are identified in the report, the above "B" paragraph should state “None”. If the report identifies no NRC-identified or self-revealing findings, Paragraph "A" should state “No findings of significance were identified”)

REPORT DETAILS

Summary of Plant Status ***(Note: Include if relevant - would be for an integrated resident report)***

Unit 1 began the period at full Rated Thermal Power (RTP) and operated at full power for the entire report period, except for a reactor trip on July 4, 2004. The unit returned to full power operation on July 11, 2004. The power on unit 1 was reduced to 65 percent power on August 9, 2004, for maintenance on the 5A feedwater heater tube side drain. Unit 1 returned to 100 percent power on August 13, 2004. ***(Note: Power reduction included because of significant duration - if only a few hours it wouldn't be worth mentioning)***

Unit 2 was in a refueling outage at the beginning of the inspection period. On July 29, 2004, Unit 2 reached full RTP and operated at or near full RTP for the remainder of the inspection period

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity ***(Note: EP is listed above the 1E section)***

1R01 Adverse Weather Protection (71111.01) (***Note: procedure number optional***)

a. Inspection Scope ***(Note: Scope describes inspection methods and references the sources for acceptance criteria)***

Since thunderstorms with potential tornados were forecast in the vicinity of the facility for August 31, 2004, the inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions. ***(Note: description of what was inspected should closely match inspection requirements.)*** On August 30, ***(Note: See IMC 0612-06.02 - "When" included because it is relevant for this inspection)*** the inspectors walked down ***(How)*** portions of the condensate system, the emergency service water (ESW) system, the ultimate heat sink, and switch yard. ***(What/where)*** These systems were selected because their safety related functions could be affected by adverse weather. The inspectors observed plant conditions, evaluating those conditions using criteria documented in NAP-00-0019, Rev. 2, "Winter Operation Preparations and Severe Weather Operation." ***(Note: Criteria for acceptability.)*** The inspectors also toured the plant grounds for loose debris, which could become missiles during a tornado, and ascertained operator staffing and if they could access controls and indications for those systems required for safe control of the plant. The documents reviewed during this inspection are listed in the Attachment. ***(Note: List documents in Attachment when more than 6 were reviewed)*** This inspection satisfied one inspection sample for the onset of adverse weather. ***(Note: include clear description and number of samples.)***

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment ***(Note: Scope should be a complete but <u>concise</u> listing of the required IP activities)***

.1 Partial Walkdown (4 samples - note: stating number of samples after the title is optional)

a. Inspection Scope

The inspectors performed a partial walkdown of the following four systems to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, walked down control systems components, and verified that selected breakers, valves, and support equipment were in the correct position to support system operation. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program. Documents reviewed are listed in the attachment.

- Unit 2 Residual Heat Removal (RHR) Train B During Mid-Cycle Train A Outage
- Emergency Gas Treatment System (EGTS) Train A during Maintenance on B Train
- Component Cooling System (CCS) Train A during Valve Testing on B Train Heat Exchanger Outlet
- Unit 1 RHR Train A During Mid-Cycle Train B Outage

.2 Complete Walkdown

a. Inspection Scope

The inspectors conducted one complete walkdown of the Unit 1 essential chilled water (ECW) system to verify the functional capability of the system. The inspectors used the licensee procedures and other documents listed below to verify proper system alignment:

- Drawing Nos. 1X4DB221, 233, and 234, Unit 1 Essential Chilled Water System
- Procedure 11744-1, Essential Chilled Water System Alignment
- Procedure 14553-1, ESF Room Cooler and Safety Related Chiller Flow Path

The inspectors also verified ECW electrical power requirements, operator workarounds, labeling, hangers and support installation, and associated support systems status. Operating pumps were examined to ensure that any noticeable vibration was not excessive, pump leakoff was not excessive, bearings were not hot to the touch, and the pumps were properly ventilated. The walkdowns also included evaluation of system piping and supports against the following considerations:

- Piping and pipe supports did not show evidence of water hammer~~.~~
- Oil reservoir levels appeared normal~~.~~
- Snubbers did not appear to be leaking hydraulic fluid~~.~~
- Hangers were functional
- Component foundations were not degraded

A review of outstanding maintenance work orders was performed to verify that the deficiencies did not significantly affect the ECW system function. The inspectors reviewed Design Change Package 99-VAN0044, which replaced certain ECW control valves and actuators, to ensure that the system design function and alignment were not adversely impacted by the changes. In addition, the inspectors reviewed the condition report (CR) database to verify that ECW equipment alignment problems were being identified and appropriately resolved

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Protection - Tours

a. Inspection Scope

The inspectors conducted a tour of the nine areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that combustibles and ignition sources, were controlled in accordance with the licensee’s administrative procedures; fire detection and suppression equipment was available for use; that passive fire barriers were maintained in good material condition; and that compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with the licensee’s fire plan. Documents reviewed are listed in the attachment.

- Emergency Diesel Generator (EDG) Building
- Control Building Elevation 669 (Mechanical Equipment Room, Battery Rooms, and Battery Board Rooms)
- Essential Raw Cooling Water (ERCW) Building
- Control Building Elevation 706 (Spreading Room)
- Auxiliary Building Elevation 714 (Corridor)
- Control Building Elevation 685 (Auxiliary Instrument Rooms)
- Auxiliary Building Elevation 690 (Corridor)
- Control Building Elevation 734 (Shutdown Board Rooms and Battery Board Rooms)
- Auxiliary Building Elevation 653 (Corridor, RHR and Containment Spray Pump Rooms)

b. Findings

No findings of significance were identified.

.2 Fire Protection - Drill Observation

a. Inspection Scope

The inspectors observed three fire drills conducted in the emergency diesel generator building and turbine building on July 15th, August 15th, and September 15th. The drills were observed to evaluate the readiness of the plant fire brigade to fight fires. These additional drills were observed to verify that the fire brigade deficiencies documented in IR 05000390,391/200404, Section 1R04, were promptly corrected. The inspectors verified that the licensee staff identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient fire fighting equipment brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) search for victims and propagation of the fire into other plant areas; (7) smoke removal operations; (8) utilization of pre-planned strategies; (9) adherence to the pre-planned drill scenario; and (10) drill objectives.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk-important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analysis and design documents, including the updated final safety analysis report (UFSAR), engineering calculations, and abnormal operating procedures, for licensee commitments. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding due to the Fire Protection System. The inspectors performed a walkdown of the fire water header in the Auxiliary Building to verify its configuration and reviewed results of the latest (July 2004) Auxiliary Building High Pressure Fire Suppression System flow test to verify that the acceptance criteria were met.

The inspectors also reviewed the licensee's corrective action documents with respect to flood-related items identified in Problem Evaluation Reports (PERs) written from January 1 through August 25, 2004, to verify the adequacy of the corrective actions. The most significant reviewed PERs written with respect to internal flooding during the period are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 External Flooding

a. Inspection Scope

The inspectors reviewed the design, material condition, and procedures for coping with the design basis probable maximum flood. First, the inspectors reviewed the flooding sections of the UFSAR to determine the barriers required to mitigate the flood. Next, the inspectors reviewed piping layout drawings and walked down the manholes for underground piping to ensure that the emergency raw cooling water (ERCW) system would remain available following the probable maximum flood. As part of this review, the inspectors also reviewed the licensee analysis for the use of cable insulation degradation due to moisture in the manholes.

The inspectors also reviewed the abnormal operating procedure (AOP) for mitigating the design basis flood. This procedure included different sections for different operating modes, however, for this review, the inspectors focused on flood mitigation with both units operating at 100% rated thermal power (RTP). The flooding AOP also included provisions for installing spool pieces in different sections of piping throughout the plant. In order to verify that these pieces were properly staged the inspectors walked down the fuel pool cooling heat exchangers, the component cooling heat exchangers, and associated ERCW piping. The inspectors also walked down the auxiliary charging system to verify that the installed equipment matched that assumed in the procedure and that the procedure would properly put the system in service.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee’s program for maintenance and testing of risk-important heat exchangers in the ERCW system. Specifically, the review included the program for testing and analysis of the B MCR chiller condenser (heat exchanger) which was cleaned, inspected, and evaluated by WO 04-812811-000 in parallel with WO 02-017913-000 to replace the condenser tubes. The inspectors observed the physical condition of the heat exchanger during the cleaning activities and verified that the frequency of inspection was sufficient to detect degradation prior to loss of heat removal capabilities below design requirements; that the inspection results were appropriately categorized against pre-established engineering acceptance criteria, including the impact of tubes plugged on the heat exchanger performance; and that the licensee had developed adequate acceptance criteria for bio-fouling controls. Additional documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

.1 Biennial Review

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of November 15-19, 2004, the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of simulator operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR 55, Operators' Licenses. The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, Operator Licensing Examination Standards for Power Reactors, and Inspection Procedure 71111.11, Licensed Operator Requalification Program. The inspectors also reviewed and evaluated the licensee's simulation facility for adequacy and for use in operator licensing examinations. The inspectors observed one operator crew during the performance of the operating tests. Documentation reviewed included written examinations, job performance measures (JPMs), simulator scenarios, licensee procedures, on-shift records, licensed operator qualification records, watch standing and medical records, simulator modification request records and performance test records, the feedback process, and remediation plans. The records were inspected against the criteria listed in Procedure 71111.11. Documents reviewed during the inspection are listed in the attachment.

Following the completion of the annual operating examination testing cycle, which ended on December 11, 2004, the inspectors reviewed the overall pass/fail results of the biennial written examination, the individual JPM operating tests, and the simulator operating tests administered by the licensee during the operator licensing requalification cycle. These results were compared to the thresholds established in Manual Chapter 609, Appendix I, Operator Requalification Human Performance Significance Determination Process.

b. Findings

Introduction: The inspectors identified a Green (Severity Level IV) non-cited violation (NCV) of 10 CFR 50.9 for failure to provide complete and accurate information for one licensed operator on his initial license application.

Description: The NRC's requirements related to the conduct and documentation of medical examinations for operators are contained in Subpart C, Medical Requirements, of 10 CFR Part 55, Operators' Licenses. Specifically, Section 55.21, Medical Examination, requires every operator to be examined by a physician when he or she first applies for a license. The physician must determine whether the operator meets the requirements of Section 55.33(a)(1), i.e., the operator's medical condition and general health will not adversely affect the performance of assigned operator duties or cause operational errors that endanger public health and safety.

Every time an operator applies for a license pursuant to Section 55.31, How to Apply, or Section 55.57, Renewal of Licenses, an authorized representative of the facility licensee must complete and sign NRC Form 396, Certification of Medical Examination by Facility Licensee, attesting, pursuant to Section 55.23, Certification, that a physician has conducted the required medical examination and determined that the operator's medical condition and general health meet the requirements of Section 55.33(a)(1). The facility licensee must also certify which industry standard (i.e., the 1983 or 1996 version of ANSI/ANS-3.4, Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants, or other NRC-approved method) was used in making the fitness determination.

The ANSI standards describe a number of specific operator health requirements and disqualifying conditions. If an operator's health does not meet the minimum standards, the facility licensee must request a conditional license in accordance with Section 55.23(b) by submitting the appropriate medical evidence with NRC Form 396. Pursuant to Section 55.33, Disposition of an Initial Application, and Section 55.57, as applicable, the Commission will review the license application based on the facility's licensee certification and include any conditions in the license that might be necessary based on the supporting medical evidence.

During the medical records review of the inspection on November 17, 2004, the inspectors determined that this operator's record indicated a need to wear corrective lenses to meet the ANSI/ANS 3.4 1983 visual acuity requirements. The facility licensee was informed that the individual required an amendment to his license that required him to wear corrective lenses while performing licensed duties. The inspectors also determined that the original NRC Form 396 submitted with his application for a license did not contain a recommendation for the no-solo license restriction that the NRC had placed on the license. In addition, a review of other medical records indicated that some operators had not taken vision tests with their corrective lenses removed and, therefore, no real baseline information existed for the actual need for corrective lenses to meet the visual acuity standard.

The inspectors reviewed the operator's docket file and determined that the facility licensee had submitted his application for a reactor operator license on November 5, 2002, which contained an NRC Form 396 signed by the site vice president certifying that the information on the document was true and correct. The form was sent with the recommendation of no restrictions on the applicant's license but had an attachment which listed prescribed medications that the applicant was taking. Region II examiners completed the administration of an initial license examination at the Dirojac Nuclear Plant in December 2002 and a license was issued on January 9, 2003 with a no-solo restriction. This restriction was imposed by the NRC and was not based on the facility licensee's certification of NRC Form 396 but on the medical information concerning prescribed medications that had been submitted as supplemental information with the NRC Form 396. Additional amplifying medical information was submitted to the NRC in a letter dated February 27, 2003, in response to the imposed no-solo condition, providing more information for the NRC medical doctor to review. The NRC then issued an amendment on March 31, 2003, which changed the wording of the no-solo restriction on his license. No information regarding the need for corrective lenses had been communicated in any of the submitted documents.

Analysis: The inspectors determined that the licensee's failure to provide complete and accurate information to the NRC, which resulted in an incorrect licensing action, is a performance deficiency because the licensee is expected to comply with 10 CFR 50.9 and because it was within the licensee's ability to foresee and prevent. Because violations of 10 CFR 50.9 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process. The finding was more than minor because information was provided to the NRC signed under oath by the site vice president which erroneously impacted an NRC licensing decision. There was no evidence that the operator endangered plant operations as a result of impaired visual acuity while performing licensed duties since the original issuance of his license in January 2002

Enforcement: 10 CFR 50.9 states, in part, "Information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects. 10 CFR 55.23 requires that an authorized representative of the facility licensee shall certify the medical fitness of an applicant by completing and signing an NRC Form 396. NRC Form 396, when signed by an authorized representative of the facility licensee, certifies that a physician conducted a medical examination of the applicant as required in 10 CFR 55.21, and that the guidance contained in ANSI/ANS 3.4-1983 was followed in conducting the examination and making the determination of medical qualification. Contrary to this, on November 5, 2002 a senior licensee representative submitted NRC Form 396 for one individual applying for a reactor operator license that certified that the applicant met the medical requirements of ANSI/ANS 3.4-1983 and that the applicant would not require any restrictions to his license. In fact, the applicant had a pre-existing medical condition and an additional requirement of corrective lenses to meet the medical standards, both of which required restrictions on his license. This information was material to the NRC because the NRC relied on this certification to determine whether the applicant met the requirements to operate the controls of a nuclear power plant pursuant to 10 CFR Part 55. The finding is not suitable for significance determination process (SDP) evaluation, but has been reviewed by NRC management and is determined to be a green finding of very low safety significance. Because the failure to provide the information requesting appropriate restrictions on the operator's license was of very low safety significance and has been entered into the corrective action program as PER 72386, this violation is being treated as a Severity Level IV non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000998/2004007-01, Failure to Provide Complete and Accurate Information to the NRC which Impacted a Licensing Decision.

The licensee took prompt corrective action and submitted a letter dated November 19, 2004, requesting a corrective lens restriction for the operator's license. The licensee entered this issue into their corrective action program, PER 72386, and conducted a 100 percent review of all operator medical records to ensure no other discrepancies existed. No other discrepancies were found.

.2 Resident Inspector Quarterly Review

a. Inspection Scope

On December 10, 2004, the inspectors observed operators in the plant's simulator during licensed operator annual requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with procedures TRN-1, Administering Training, and TRN-11.4, Continuing Training for Licensed Personnel. The inspectors observed a shift crew's response to the two scenarios listed below:

- 3-OT-SRE0005, Main Steamline Break Inside Containment/Steam Generator Tube Rupture
- 3-OT-SRE0002, Reactor Trip with Steam Generator Safety Valve Failed Open

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two samples listed below for items such as: (1) appropriate work practices; (2) identifying and addressing common cause failures; (3) scoping in accordance with 10 CFR 50.65(b) of the maintenance rule (MR); (4) characterizing reliability issues for performance; (5) trending key parameters for condition monitoring; (6) charging unavailability for performance; (7) classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2); and (8) appropriateness of performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified as (a)(1). In addition, the inspectors specifically reviewed events where ineffective equipment maintenance has resulted in invalid automatic actuations of Engineered Safeguards Systems affecting the operating units. Documents reviewed are listed in the Attachment. Items reviewed included the following:

- Snubber Failures and Maintenance. The failures were documented as part of the licensee's corrective action program in the following PERs: 69448, 47817, 44427, 47623, 41692, 48020, 61924, 71278, 44318, 44457

- Safety-Related Breaker Performance

b. Findings

Introduction: A Green inspector-identified NCV of 10 CFR 50.65, Requirements for monitoring the effectiveness of maintenance at nuclear power plants, was identified for the licensee's failure to demonstrate that the performance or condition of Reactor Motor-Operated Valve (RMOV) Board 1B was effectively controlled through appropriate preventive maintenance. As a result, the licensee did not establish goals or monitor the performance of the board per 10 CFR 50.65a(1) to ensure that appropriate corrective actions were initiated.

Description: The inspectors reviewed PER and WO records related to the loss of the safety-related breakers. The inspectors noted that the feeder breaker for RMOV board 1B had de-energized three times between August 26, 2003 and June 23, 2004. In each case, a load was being started but the individual load breaker did not trip open. In a typical selective trip design, the load breaker should trip open and not affect the feeder breaker to the board. The loads involved were the Unit 1 Reactor Protection System Bus (RPS) MG set B motor, the Control Bay Supply Fan motor 1B and RHRSW sump pump B motor in pump compartment C, respectively. (See additional details on this load in Section 4OA2). When this board trips, Reactor Protection System 1B de-energizes and the Standby Gas Treatment and Control Room Emergency Ventilation systems receive an automatic start signal. Plant operators on the operating units are required to respond to the annunciators associated with the unexpected start of these systems, assess plant conditions, and then realign the systems back to their normal standby configuration. Though some WO's were initiated and some breaker subcomponents were replaced, and the board's normal feeder breaker and alternate feeder breaker have been exchanged, no cause has yet been determined. At the conclusion of this inspection, there were outstanding work orders dating back to April of 2004. This board primarily affects systems on the non-operating Unit 1. However, the RHRSW sump pump B is common plant equipment, is safety-related, and is designed to protect other safety related common equipment. In addition, common Engineered Safety Feature equipment automatically starts in response to these equipment problems.

The inspectors reviewed licensee procedure 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting-10 CFR 50.65, and noted that the functional failure criteria for this system is the loss of a 480-V board for more than five minutes. The performance criteria is no more than one functional failure per Unit in a 24-month rolling period. The inspectors' review of the operating logs indicated that the board was de-energized on April 22, at 14:06 and was re-energized at 15:00 for a total of 54 minutes and on June 23, the board was de-energized at 09:00 and re-energized at 10:00, for a total of 60 minutes. These two functional failures placed the board (System 268) in the 10 CFR 50.65(a)(1) category for not meeting the performance criteria. However, the licensee had not accounted for these functional failures and out-of-service times or identified that the board (system 268) had not met their performance criteria. The licensee had not established any additional performance monitoring goals or identified specific corrective actions. The licensee entered the problem with their failure to monitor the performance of the RMOV into their CA program for resolution.

Analysis: The inspectors determined that the licensee's failure to demonstrate that the performance or condition of the RMOV Board 1B was capable of achieving its specified reliability criteria was more than minor because it affected the reliability objective of the Equipment Performance attribute under the Mitigating Systems Cornerstone. The inspectors assessed the finding using the SDP and determined the finding to be of very low safety significance. The finding was of low safety significance because there was no design deficiency and the equipment affected by the board failure either failed in a safe manner or had its redundant equipment functional.

Enforcement: 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Paragraph (a)(2) states, "Monitoring as specified in paragraph (a)(1) of this section is not required where it has been demonstrated that the performance or condition of a structure, system, or component is being effectively controlled through the performance of appropriate preventive maintenance, such that the structure, system, or component remains capable of performing its intended function." Paragraph (a)(1) states, in part, that the licensee "...shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components...are capable of fulfilling their intended functions." Contrary to the above, prior to June 23, 2004, the licensee failed to demonstrate that the performance or condition of RMOV Board 1B was being effectively controlled through the performance of appropriate preventive maintenance such that the system remained capable of performing its intended function. Therefore, between June 23, 2004, and December 30, 2004, the licensee failed to establish goals and monitor RMOV Board 1B under paragraph a(1) or demonstrate that monitoring under a(1) was not required. The failure is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000998,999/2004007-02: Failure to Demonstrate that the RMOV Board 1B Performance Was Effectively Controlled per 10 CFR 50.65 (a)(2). This issue is in the licensee's Corrective Action Program as PER 74450.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following six activities to verify that the appropriate risk assessments were performed prior to removing equipment for work. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors verified the appropriate use of the licensee's risk assessment tool and risk categories in accordance with Procedure SPP-7.1, On-Line Work Management, Revision SS1, and Instruction 0-TI-DSM-000-007.1, Risk Assessment Guidelines, Revision 8. Documents reviewed are listed in the attachment.

- RHR Pump 1A failure to start during surveillance
- Unit 2 RHR, Containment Spray, and Safety Injection A Train Outage
- Unit 2 Centrifugal Charging Pump B Train Outage
- Replacement of 6.9-kV Auto-Close Siemens Breakers with ABB Breakers
- Unit 1 Component Cooling Train B Outage
- Unit 1 Auxiliary Building Ventilator and Electric Board Room Chiller A out-of-service concurrently

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Plant Evolutions and Events

a. Inspection Scope

For the non-routine events described below, the inspectors reviewed operator logs, plant computer data, and strip charts to determine what occurred and how the operators responded, and to determine if the response was in accordance with plant procedures:

- On January 18, 2004, the inspectors observed the site response to a trip of the Unit 1 reactor building "B" chiller and a failure of the "A" chiller to automatically load. Normal drywell cooling was temporarily lost and the air temperature increased to 136.6 degrees Fahrenheit (°F), which was above the Technical Specification (TS) limit of 135°F. The "A" chiller was manually loaded and drywell temperature returned to 130°F, below the TS value.

- On February 12, 2004, the inspectors observed the site response to a "D" EDG over-voltage alarm. The licensee declared the "D" EDG inoperable and implemented TS 3.8.1, "AC Sources - Operating." The inspectors observed site maintenance activities (Work Order 293407), control of plant risk, implementation of TS, and common cause failure analysis. The licensee determined that the alarm resulted from a faulty relay base in the alarm circuit and this condition would not have prevented the EDG from performing its required safety functions. The relay base was replaced and the EDG returned to service on February 23, 2004.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the six operability evaluations described in the PERs listed below, the inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors reviewed the UFSAR to verify that the system or component remained available to perform its intended function. In addition, the inspectors reviewed compensatory measures implemented to verify that the compensatory measures worked as stated and the measures were adequately controlled. The inspectors also reviewed a sampling of PERs to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the attachment.

- PER 64015, Evaluation of Auxiliary Building Room Coolers with Access Panels Removed
- PER 64477, Failure of Eagle 21 Logic Card in Unit 2 Protection Set 4, Rack 13
- PER 64674, RHR Pump 1A Failed to Start on Demand
- PER 66924, Higher-Than-Predicted Water Gap Closure on Framatome Alliance Lead Test Assemblies
- PER 61789, Nuclear Instrument N41 Upper Detector Ammeter Found Out-of-Tolerance
- PER 62486, One Section of Intake Damper for Diesel Generator 2B Failed Closed

b. Findings

Introduction: The inspectors identified a Green non-cited violation (NCV) for a failure to comply with TS 3.3.1 when a Loop Control Processor (LCP) failed in Unit 2.

Description: On July 1, 2004, an LCP failure occurred in Protection Set 4 Rack 13 of the Eagle 21 Reactor Protection System of Unit 2. This affected Delta T, Tavg, pressurizer pressure, PORV Interlock, Refueling Water Storage Tank Level, Containment Sump Level, and Wide-Range Steam Generator Level control and protection functions. Operators followed actions of the appropriate AOPs to defeat the control functions of the affected channels and attempted to reset the LCP, but were unsuccessful.
In addition, based on the licensee position that a LCP failure placed all associated channel bistables in the correct TS position for an inoperable channel, operators took no further action to trip the bistables. Approximately 9½ hours after the failure occurred, operators tripped all channel bistables when preparing to replace the failed LCP.

The inspectors reviewed logs and procedures, compared TS requirements to the actions taken, and interviewed licensee operations and engineering staff members. The inspectors also reviewed the licensee's written position on LCP failure and discussed it with the NRC Office of Nuclear Reactor Regulation. From this, the inspectors determined that operator action to trip the channel bistables 9½ hours after the failure occurred did not comply with TS 3.3.1, Action 9, and TS 3.3.2, Action 36, which required the inoperable channels to be placed in the tripped condition within six hours. The licensee entered their failure to comply with TS into their CA program for resolution

Analysis: The failure by the operators to trip the inoperable channels within six hours was more than minor because it affected the configuration control attribute of the Mitigating Systems Cornerstone in that it reduced the reliability of the required number of operable channels required by the reactor protection system. Had actual plant conditions called for a trip, not taking deliberate operator action to place the inoperable channels in a tripped condition would reduce the likelihood of proper coincident protection system actuation. The TS action statement to "place" the channel in the tripped condition is deliberate in that there is no assurance that the channel will fail in the safe condition and raises an operability question if this action is not taken. Because there was no loss of safety function and the bistables were actually in the tripped condition due to the failed LCP input, the failure to meet the TS was considered to be of very low safety significance (Green).

Enforcement: TS 3.3.1 requires that inoperable channel bistables be placed in a tripped condition within six hours. Contrary to the above, on July 1, 2004, the licensee failed to place the inoperable channel bistables for functions served by the LCP in Protection Ste 4, Rack 13 of the Dirojac Unit 2 RPS in a tripped condition within that time. Because this violation was determined to be of very low safety significance, it is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000999/2004007-03, Failure to Comply with TS 3.3.1. This violation is in the licensee's corrective action program as PER 64477.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed the cumulative effects of deficiencies that constituted operator workarounds to determine whether or not they could affect the reliability, availability, and potential for misoperation of a mitigating system; affect multiple mitigating systems; or affect the ability of operators to respond in a correct and timely manner to plant transients and accidents. The inspectors also assessed whether operator workarounds were being identified and entered into the licensee's corrective action program at an appropriate threshold. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed licensee procedures 0-TI-405, Plant Modifications and Design Change Control, and SPP-9.3, Plant Modifications and Engineering Change Control, and observed part of the licensee's activities to implement a design change, that affected all units, while the units were online. The inspectors reviewed the associated 10 CFR 50.59 screening against the system design bases documentation to verify that the modifications had not affected system operability/availability. The inspectors reviewed selected ongoing and completed work activities to verify that installation was consistent with the design control documents. Design Change Notice (DCN) 60600, Upgrade the Common Reactor Building 125-Ton Bridge Crane, was reviewed.

b. Findings

Introduction: A Green self-revealing NCV was identified for the failure to comply with 10 CFR 50 Appendix B, Criterion V, Instructions, Procedures and Drawings. As a result of an inadequate procedure and poor human performance, a Reactor Building crane trolley drop occurred.

Description: On August 24, licensee and contract personnel were conducting work using WO 04-716728-000, the implementing work document, for Design Change Notice (DCN) 60600 to upgrade the common Reactor Building 125-ton bridge crane. Part of the DCN was to replace the 65000-pound trolley with a new one. During the rigging process to remove and lower the old trolley from the overhead to the Unit 1 refueling floor, one synthetic sling failed and one end of the trolley dropped approximately four feet to the concrete floor. The force associated with the drop resulted in the failure of one of the two remaining slings on the other end of the trolley. Operations and engineering personnel immediately performed a series of detailed inspections and determined that no plant operability or safety issue resulted. The licensee determined that the event did not challenge the safe operation of Unit 2 or cause entry into any Limiting Conditions of Operation. The drop resulted in surface cracking and spalling of the concrete ceiling beneath the point of impact on the Unit 1 refueling floor. The licensee assembled a root cause investigation team to review the event and determine its root cause. The licensee also commissioned the services of an independent structural engineer to analyze the structural integrity of the floor at the point of impact to determine if the floor still met its design criteria.

The inspectors completed a walkdown of the affected areas, accompanied by a civil engineer from the licensee's staff, to view the cracked and spalled concrete from the ceiling below the point of impact. The inspectors also toured the plant and the main control rooms to assess the condition and status of safety-related systems. The inspectors discussed the issue with licensee management, engineering, and operations personnel to assess immediate actions taken and gain an understanding of the detailed inspections completed by licensee personnel. The inspectors also assessed compliance with the reporting requirements of NUREG-1022, Event Reporting Guidelines.

The inspectors later reviewed the licensee's root cause determination report to assess details, accuracy, and short and long term corrective actions. The inspectors noted that the root cause report was thorough, detailed, and comprehensive. The planned and completed actions were appropriate and comprehensive. The licensee identified several root and contributing causes. Root causes included inadequate work practices by the contractor support personnel, and improper installation and verification of the rigging in that the synthetic slings used in the lift were not adequately protected.

The inspectors compared the root and contributing causes with information obtained from the review of licensee work control documents, procedures, briefing papers listed in the attachment, and discussions with licensee personnel. The procedure to remove the old trolley and install the new trolley was revised several times prior to its implementation. However, the rigging crew was not made aware of the final revision and did not implement all of the requirements to use "softeners" to protect the slings and that a line of sight be maintained to ensure that their effectiveness was maximized.

The licensee's investigation indicated that a single sling was rigged around the trolley support beam with five protective softeners. The softeners were verified at the beginning of the move but not during the move, as specified by the rigging permit. Photographs showed that at least one softener at the trolley beam was not in a position to protect the sling after the load was applied. As the old trolley was lowered close to the new trolley, which was staged in preparation for its installation, workers were concerned about possible interference between them. The contract project lead engineer determined that there would be additional clearance if one end of the old trolley was lowered. There was no discussion or intervention by the Dirojac task manager, supervisor, or safety observer, even though at the pre-job briefing it was emphasized that the load was to be maintained level. The trolley descent had been halted several times to level the load. When one end of the trolley was lowered, the edge of the trolley beam cut the single rigging sling and the trolley fell. Almost immediately, one of the slings on the other end of the trolley failed and the trolley fell to the refuel floor.

<u>Analysis</u>: The inspectors determined that the licensee's inadequate procedure and poor work performance which resulted in the Reactor Building crane trolley drop that occurred on August 24, 2004, constituted a performance deficiency and a finding. This finding is greater than minor because it is associated with program and process attributes and affected the objective of the Initiating Event Cornerstone to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations. In addition, if left uncorrected, this finding would result in a more significant safety concern because structural damage to the refuel floor as well as potential damage to the spent fuel pool would occur if the load had dropped from a higher elevation. This finding did not represent an immediate safety concern. This finding was evaluated using the SDP and was determined to be a finding of very low safety significance because no initiating event or transient actually occurred, there was

no permanent structural damage to the refuel floor, there was no functional degradation, and mitigating capability was not affected. The inspectors also determined that the cause of this finding was related to the personnel aspect of the human performance cross-cutting area.

Enforcement: 10 CFR 50 Appendix B, Criterion V, states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, inadequate procedures (not using the latest approved revision to the procedure) and poor human performance resulted in the drop of the Reactor Building crane on August 24, 2004. Because this failure to comply with 10 CFR 50, Appendix B, Criterion V, is of very low safety significance and has been entered into the licensee's corrective action program, as PER 70752, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000998, 999/2004007-04, Inadequate Procedure and Poor Human Performance Resulted in a Drop of the Reactor Building Crane Trolley.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the seven post-maintenance tests listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the licensee's test procedure to verify that the procedure adequately tested the safety function(s) that may have been affected by the maintenance activity, that the acceptance criteria in the procedure were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). Documents reviewed are listed in the attachment.

- Work Order (WO) 04-771681-000, Charging Pump 2B Train Outage
- WO 02-004750-000 and WO 02-004750-003, MCR Chiller B Oil Leaks and Compressor Replacement
- WO 04-775100-000, EDG 2B Idle Speed Testing and Relay Replacement
- WO 04-778943-000, Replace 30RX and 1X Relays in Control Circuit for Containment Spray 1B Motor
- WO 03-012491-000, Rebuild ERCW Pump R-A
- WO 04-779355-000, Containment Spray Pump 2B Breaker Cell Switch Repair and Adjustment
- WO 03-014194-000, MOVAT Testing on RHR 1B Minimum Flow Control Valve

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the Unit 2 refueling outage, conducted March 2 - 25, 2004, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the Attachment.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out of service.

- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.

- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and an accounting for instrument error.

- Controls over the status and configuration of electrical systems to ensure that TS and outage safety plan requirements were met, and controls over switchyard activities.

- Monitoring of decay heat removal processes.

- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.

- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.

- Controls over activities that could affect reactivity.

- Maintenance of secondary containment as required by TS.

- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.

- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.

- Licensee identification and resolution of problems related to refueling outage activities.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed six surveillance tests and/or reviewed test data of selected risk-significant SSCs, listed below, to assess, as appropriate, whether the SSCs met the requirements of the TS; the UFSAR; SPP-8.0, Testing Programs; SPP-8.2, Surveillance Test Program; and SPP-9.1, ASME Section XI. The inspectors also determined whether the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. Additional documents reviewed are listed in the attachment.

- WO 04-815559-000, 0-SI-65-8-B, Emergency gas treatment system filter Train B test
- WO 04-815564-000, 0-SI-30-9-B, Auxiliary building gas treatment system filter Train B test
- WO 04-818471-000, Perform 0-SI-82-12-B, Monthly diesel generator start and load test DG 2B-B
- WO 04-817480-000, Perform 1-SI-3-901-B, Motor-driven auxiliary feedwater pump 1B-B quarterly performance test
- WO 04-819346-000, Perform 1-SI-3-902, Turbine-driven auxiliary feedwater pump 1A-S quarterly performance test
- 1-SI-0-4, Monthly Surveillances

b. Findings

Inadequate Procedure for Surveillance of Remote Shutdown System Instrumentation

Introduction: A Green NCV of T.S. 5.7.1.1.a was identified by the NRC regarding an inadequate procedure for surveillance of remote shutdown system instrumentation.

Description: On September 15, 2004, the inspectors identified that remote shutdown instrumentation, 1-TI-68-65C (hot leg loop 4 temperature) exceeded the maximum channel deviation (MCD) in that the instrument was at top-of-scale (650 degrees Fahrenheit [°F] and apparently failed high. TS 3.3.4, Remote Shutdown System (RSS), identifies required instrumentation for unit shutdown in the event of conditions forcing the evacuation of the normal control room and is considered to be an important contributor to the reduction of unit risk to accidents. TS Surveillance Requirement (SR) 3.3.4.1 requires an instrument channel check every 31 days. This is implemented by 1-SI-0-4, Monthly Surveillances, and consists of a comparison of the instrument located in the auxiliary control room to the equivalent instrument in the main control room to determine if the difference between channels is within a band defined as the MCD, e.g., 30 °F for the reactor coolant hot legs temperature instrumentation. Any deviation beyond this results in a declaration of inoperability of the affected instrument channel.

The licensee evaluated the condition, declared the instrument inoperable, and entered a 30-day action statement. The inspectors reviewed the previous performance of 1-SI-0-4 on September 4, 2004, and observed that 1-TI-68-65C was recorded as 650 °F versus a control room reading (1-TI-68-65) of 620 °F with no corrective action initiated. The inspectors also determined that 1-SI-0-4 did not have instructions for operator response when the MCD encompasses the top-of-scale or a failed-high indication for a particular instrument. Therefore, a condition of undetected inoperability is possible and did exist on September 15, 2004, as discovered by the inspectors. The inadequate establishment and maintenance of this procedure is contrary to TS 5.7.1.1.a, which requires that written procedures be established, implemented, and maintained as specified in RG 1.33, Revision 2, of which Appendix A, Item 8b, states that implementing procedures are required for each surveillance test listed in the TS. The licensee entered this performance deficiency into their CA program for resolution.

<u>Analysis</u>: The inspectors referred to MC 0612 and determined that the finding is greater than minor in that it affected the ability of the licensee to monitor the status of the reactor following a control room evacuation and is associated with the Mitigating Systems cornerstone and the respective attribute of procedure quality. The inspectors evaluated this finding using MC 0609 and determined that it was of very low safety significance (Green) because it did not result in a loss of function per Generic Letter (GL) 91-18, did not represent an actual loss of safety function, and was not potentially risk-significant due to possible external events. A contributing cause of the finding is related to the cross-cutting element of human performance, in that operators did not identify that the instrument was recorded as being at it's maximum channel deviation when the TS surveillance was performed 10 days earlier.

<u>Enforcement</u>: TS 5.7.1.1.a requires that written procedures be established, implemented, and maintained for the activities specified in RG 1.33, Revision 2, Appendix A. Item 8b of RG 1.33 states that implementing procedures are required for each surveillance test listed in the TS. Contrary to this, 1-SI-0-4, Monthly Surveillances, was not adequately established or maintained and, consequently on September 4, 2004, the inoperability of 1-TI-68-65C (hot leg loop 4 temperature) was not identified. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program as PER 68838, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000998/2004007-05,

Inadequate Procedure for Surveillance of Remote Shutdown System Instrumentation.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the two temporary modifications listed below and the associated 10 CFR 50.59 screening, and compared each against the UFSAR and TS to verify that the modification did not affect operability or availability of the affected system. The inspectors walked down each modification to ensure that it was installed in accordance with the modification documents and reviewed post-installation and removal testing to verify that the actual impact on permanent systems was adequately verified by the tests.

- TACF 1-04-0019-067, Leak Repair of Tube Leak on 1A Spent Fuel Pump/Thermal Barrier Booster Pump Area Cooler
- TACF 0-04-026-032, Temporary Compressor for Station Control and Service Air

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

Resident inspectors evaluated the conduct of a routine licensee emergency drill on August 18, 2004 to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Plan Classification Matrix, Revision 35. The inspectors also attended the licensee critique of the drill to compare any inspector-observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying failures.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically-Significant Areas

a. Inspection Scope

This area was inspected to assess the licensee's performance in implementing physical and administrative controls, including worker adherence to these controls, for airborne radioactivity areas, radiation areas, high radiation areas, and very high radiation areas. The inspectors used the requirements in 10 CFR Part 20, the Technical Specifications, and the licensee's procedures required by the Technical Specifications as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator (PI) events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone

- Controls (surveys, postings, and barricades) of three radiation, high radiation, and airborne radioactivity areas

- Radiation work permit procedure, engineering controls, and air sampler locations

- Conformity of electronic personal dosimeter alarm setpoints with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms

- Barrier integrity and performance of engineering controls in two potential airborne radioactivity work areas

- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within the spent fuel storage pool

- Self-assessments and audits related to the access control program since the last inspection

- Corrective action documents related to access controls

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions

- Adequacy of radiological controls such as required surveys, radiation protection job coverage, and contamination controls during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

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

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirems committed effective dose equivalent

- Licensee event reports (LERs) and special reports related to the access control program since the last inspection

The inspectors completed 21 of the required 21 samples.

b. Findings

Introduction. A Green, self-revealing NCV of Technical Specification 5.7.3 was identified. Three workers were exposed to unanticipated radiation levels of approximately 1,700 millirems per hour because the licensee's radiation protection technicians failed to identify and control an existing high radiation area with dose rates greater than 1,000 millirems per hour in the drywell.

Description. On July 31, 2004, three workers entered the drywell to perform maintenance activities on valves located on the 82-foot elevation. The three workers' electronic alarming dosimeters unexpectedly alarmed when they were exposed to unanticipated radiation levels of approximately 1,700 millirems per hour. Subsequent surveys at the source of radiation around Valve RCS-V-3009 measured 6,000 millirems per hour on contact and 2,000 millirems per hour at 30 centimeters. The area was not barricaded or conspicuously posted. It was not practical to lock the area; however, it did not have a flashing light activated as a warning device. The licensee determined that the three workers received 84, 85, and 95 millirems, respectively.

Analysis. The failure to control access to a high radiation area is a performance deficiency. The finding is more than minor because it is associated with the occupational radiation safety cornerstone attribute of exposure control and affected the cornerstone objective, because not controlling locked high radiation areas could increase personal exposure.

Since this occurrence involved workers' unplanned, unintended dose or potential for such a dose that could have been significantly greater as a result of a single minor, reasonable alteration of circumstances, this finding was evaluated with the occupational radiation safety significance determination process. The inspectors determined that the finding was of very low safety significance (Green) because it did not involve (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding was entered into the licensee's corrective action program.

Enforcement. Technical Specification 5.7.3.a states, in part, that for individual high radiation areas with radiation levels greater than or equal to 1,000 millirems per hour that are accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that is not continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device. On July 31, 2004, the licensee violated this requirement when it did not properly control the high radiation area with dose rates greater than 1,000 millirems per hour.

Because the failure to control a high radiation area was determined to be of low safety significance (Green) and was entered into the licensee's corrective action program as CR-RBS-2004-03551, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy, NUREG-1600 (NCV 05000998/2004007-08).

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

a. Inspection Scope

Radiation Monitoring Instrumentation: During tours of the auxiliary building and refueling floor, the inspectors observed installed radiation detection equipment including the following instrument types: Area Radiation Monitors (ARMs), Continuous Air Monitors (AMS-4s), Personnel Contamination Monitors (PCM-1Bs), and components of the Post-Accident Sampling System (PASS). The inspectors observed the physical location of the components, noted the material condition, and compared sensitivity ranges with the Updated Final Safety Analysis Report (UFSAR) and other applicable requirements. The inspectors also observed HP technicians' use of portable air samplers and survey meters during an at-power entry into U1 upper containment.

In addition to equipment walk-downs, the inspectors observed functional checks and alarm setpoint testing of various fixed and portable detection instruments.

These observations included: calibration of a refueling floor ARM; response checks of portable ion chambers and teletectors; and source checks of electronic dosimeters and a whole body counter. The most recent 10 CFR Part 61 analysis for Dry Active Waste (DAW) was reviewed to determine if calibration and check sources are representative of the plant source term.

The inspectors reviewed the two most recent calibration records for an auxiliary building AMS-4 and for all U2 containment high-range ARMs. The records were evaluated to determine frequency and adequacy of the calibrations. In addition, calibration stickers on portable survey instruments were noted during inspection of storage areas for “ready-to-use” equipment.

Operability and reliability of selected radiation detection instruments were reviewed against details documented in the following: 10 CFR Part 20; NUREG-0737, Clarification of TMI Action Plan Requirements; TS Section 3; UFSAR Chapter 12; and applicable licensee procedures. Documents reviewed during the inspection are listed in Section 2OS3 of the report attachment.

<u>Self-Contained Breathing Apparatus (SCBA) and Protective Equipment</u>: Selected SCBA units staged for emergency use in the Control Room and other locations were inspected for material condition, air pressure, and number of units available. The inspectors also reviewed maintenance records for components of four SCBA units for the past five years and certification records associated with supplied air quality.

Qualifications for off-site staff (no maintenance is performed on-site) responsible for testing and repairing SCBA equipment were evaluated through review of training records. In addition, three Control Room operators were interviewed to determine their knowledge of available SCBA equipment locations, including corrective lens inserts if needed, and their training on bottle change-out during a period of extended SCBA use. Respirator qualification records were reviewed for several Control Room and emergency response (fire brigade) personnel.

Licensee activities associated with maintenance and use of respiratory protection equipment were reviewed against 10 CFR Part 20; Regulatory Guide (RG) 8.15, Acceptable Programs for Respiratory Protection; ANSI-Z88.2-1992, American National Standard for Respiratory Protection; and applicable licensee procedures. Documents reviewed during the inspection are listed in Section 2OS3 of the report attachment.

<u>Problem Identification and Resolution</u>: Three licensee PERs and one Self-Assessment associated with instrumentation and protective equipment were reviewed and assessed. The inspectors evaluated the licensee’s ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure SPP-3.1, Corrective Action Program, Rev. 7S1. Documents reviewed are listed in Section 2OS3 of the report Attachment.

b. Findings

No findings of significance were identified.

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

Radioactive Effluent Treatment and Monitoring Systems: The operability, availability, and reliability of selected effluent process sampling and detection equipment were reviewed and evaluated. Inspection activities included record reviews and direct observation of equipment installation and operation. Current calibration data were reviewed for the selected process monitors.

The inspectors reviewed the most current Radioactive Effluent Report to assess report content and program implementation for consistency with TS, Offsite Dose Calculation Manual (ODCM) requirements and the guidance in RG 1.21, "Measuring, evaluating and reporting radioactivity in solid wastes, and releases of radioactive materials in liquid and gaseous effluents from light-water cooled nuclear power plants." Changes to the current ODCM were also evaluated.

The accessible major components of the gaseous and liquid effluent processing and release systems were observed for material condition and for system configuration with respect to descriptions in the UFSAR and ODCM. Material condition, operability, and alarm set points were assessed for five effluent radiation monitoring systems. The inspectors assessed whether compensatory sampling and analyses were performed as required when effluent monitors were out of service. Calibration records for five effluent radiation monitors, one count room gamma spectroscopic instrument, and one liquid scintillation instrument were reviewed to assess whether required surveillances were current and whether procedurally established acceptance criteria were met. The selected process monitors were associated with liquid radwaste, (blowdown, sump discharge, essential raw water cooling, and cask decon collector tank) and gaseous effluents (shield building exhaust, auxiliary building vent, and containment purge). The inspectors reviewed the licensee's quality control (QC) evaluations of intra-laboratory comparison analytical results for samples typical of plant effluents.

Equipment configuration, material condition, and operation for the effluent processing, sampling, and monitoring equipment were reviewed against details documented in TS; 10 CFR Part 20; UFSAR Sections 11 and 12; ODCM, Rev. 47; American National Standards Institute (ANSI)-N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities; ANSI-N13.10-1974, Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents; and approved procedures listed in Section 2PS1 of the report Attachment.

Effluent sampling task evolutions, and offsite dose results were evaluated against 10 CFR Part 20 requirements, Appendix I to 10 CFR Part 50 design criteria, TS, UFSAR details, ODCM, and applicable procedures listed in Section

2PS1 of the attachment. Laboratory QC activities were evaluated against RG 1.21, Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials In Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plant, June 1974; and RG 4.15, Quality Assurance for Radiological Monitoring Programs (Normal Operation) - Effluent Streams and the Environment, December 1977.

Problem Identification and Resolution. Eight PERs and one audit associated with effluent processing and monitoring activities were reviewed and discussed with Chemistry personnel. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with licensee procedure SPP-3.1, CAP, Rev. 7S1. Specific documents reviewed are listed in the report attachment.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program

a. Inspection Scope

REMP Implementation: The environmental monitoring program guidance and implementation activities were inspected. The inspection consisted of direct physical observation of sample stations, sample collection, sample preparation, review of the Annual Environmental Operating Reports for 2003 and 2004 and documentation, and interviews with licensee personnel.

The inspectors observed the routine weekly collection of five airborne particulate and iodine samples and the collection of a milk sample. The observed sample collection locations were LM-2, PM-9, PM-2, PM-3, and RM-2. The inspectors observed the material condition of one water composite sampler at the City of Dayton Municipal Water Intake, five air samplers and five co-located rainfall composite sampling devices. Milk collection from a local dairy farm, Farm HW, was observed. Environmental thermoluminescent dosimeters (TLDs), in the vicinity of the air sampling stations, were checked for material condition and appropriate identification. TLDs examined included WSW-2A, SW-2, W-3, –4, NW-2, NNW-3 and co-located TLDs: Dirojac NNE-4 and Dirojac SW-3.

Air flow calibration records were reviewed for sampler numbers LM-2 and PM-9. The inspectors independently determined the sampling locations using a handheld global positioning system (GPS) instrument. The inspectors compared the GPS locations with licensee measurements, the ODCM specified locations, and the Annual Radiological Environmental Operating Report.

Results of inter-laboratory comparisons for typical REMP sample types during calendar year (CY) 2003 and 2004 were reviewed and evaluated.

Licensee procedures and activities related to environmental monitoring were evaluated for consistency with the TS, UFSAR, and ODCM. The licensee's environmental monitoring related procedures, reports and records reviewed during the inspection are listed in Section 2PS3 of the report Attachment.

Meteorological Monitoring Program: The inspectors walked down the meteorological tower and its supporting instrumentation and observed the physical condition of the equipment. The inspectors compared system-generated data with the data provided by the plant computer to various locations including the control room. The data were also compared with the inspectors' observations of wind direction and speed measured at the tower. The inspectors also assessed system reliability and data recovery. Meteorological tower siting was evaluated based on near field obstructions, ground cover, proximity to the plant, and distance from terrain that could affect the representativeness of the measurements. The inspectors reviewed the calibrations and trouble reports for selected meteorological tower sensors used during the previous year.

Licensee procedures and activities related to meteorological monitoring were evaluated for consistency with TS, ODCM, UFSAR Section 2.3 Meteorology, and ANS/ANSI 3.11-2000, Determining Meteorological Information at Nuclear Facilities.

Unrestricted Release of Material from the RCA: Radiation protection activities associated with radioactive material control and the unconditional release of materials from the RCA were reviewed and evaluated. The inspectors observed surveys of personnel and material being released from the RCA and evaluated licensee response to detector alarms. Functional source checks using Gamma Tool Monitor (GTM), personnel contamination monitor (PCM-1B), and gamma-sensitive portal monitor (PM-7) equipment were observed and detector sensitivity was discussed with HP supervision. To evaluate the appropriateness and accuracy of release survey instrumentation, radionuclides identified within recent waste stream analyses were compared against the radionuclides used in current performance check sources. In addition, the two most recent calibration records for selected GTMs, PCM-1Bs, and PM-7s were reviewed.

Licensee programs for monitoring materials and personnel released from the RCA were evaluated against 10 CFR Part 20 and IE Circular 81-07, Control of Radioactively Contaminated Material. Licensee documents reviewed are listed in Section 2PS3 of the report attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

Cornerstone: Mitigating Systems

The inspectors sampled licensee submittals for the two PIs listed below for units one and two. For the high pressure injection unavailability and for the functional failures, the inspectors looked at the period from second quarter 2003 through the first quarter 2004.
To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

- Safety System Unavailability: High Pressure Injection System
- Safety System Functional Failures

The inspectors reviewed portions of the operations logs and raw PI data developed from monthly operating reports and discussed the methods for compiling and reporting the PIs with cognizant licensing, engineering, and maintenance rule personnel. The inspectors also independently screened maintenance rule cause determination and evaluation reports and calculated selected reported values to verify their accuracy. The inspectors compared graphical representations from the most recent PI report to the raw data to verify that the data was correctly reflected in the report. Licensee event reports (LERs) issued during the referenced time frame were also reviewed for safety system functional failures and are listed in the attachment.

Cornerstone: Occupational Radiation Safety

- Occupational Exposure Control Effectiveness

The inspectors reviewed PER records generated from June 2003 through August 2004 to ensure that radiological occurrences were properly classified per NEI 99-02. The inspectors also reviewed electronic dosimeter alarm logs, radioactive material intake records, and monthly PI reports for CY 2004. In addition, licensee procedural guidance for classifying and reporting PI events was evaluated. Reviewed documents are listed in Section 4OA1 of the report attachment.

Cornerstone: Public Radiation Safety

- RETS/ODCM Radiological Effluents Occurrence

The inspectors reviewed records used by the licensee to identify occurrences of quarterly doses from liquid and gaseous effluents in excess of the values specified in NEI 99-02 guidance. Those records included monthly effluent dose calculations for CY 2004. The inspectors also interviewed licensee personnel that were responsible for collecting and reporting the PI data. In addition, licensee procedural guidance for classifying and reporting PI events was evaluated. Reviewed documents are listed in Section 4OA1 of the report attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

(Note: Section 4OA2 is a required section for all inspection reports. This section documents: the results of PI&R reviews performed with the 10% to 15% of inspection hours allocated in each inspection procedure; the semi-annual trend review; the three to six annual samples reviewed in accordance with IP 71152 with sufficient detail in the Inspection Scope section to allow for integration into the biennial assessment; and the cross-reference of the PI&R insights associated with findings documented elsewhere in the report. The biennial team inspection conducted in accordance with IP 71152 should normally be issued in a separate inspection report.)

.1 Review of Items Entered into the Corrective Action Program:

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed screening of all items entered into the licensee's corrective action program. This was accomplished by reviewing the description of each new PER and attending daily management review committee meetings. Documents reviewed are listed in the attachment.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, the inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment and corrective maintenance issues but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.1. The review also included issues documented outside the normal CAP in system health reports, corrective maintenance WOs, component status reports, site monthly meeting reports and maintenance rule assessments. The inspectors' review nominally considered the six-month period of June through December 2003, although some examples expanded beyond those dates when the scope of the trend warranted. The inspectors compared and contrasted their results with the results contained in the licensee's latest integrated quarterly assessment report. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy. The inspectors also evaluated the trend report specified in SPP-3.1, Corrective Action Program, and 10 CFR 50 Appendix B. Specific documents reviewed are listed in the attachment.

b. Assessment and Observations

The inspectors identified four issues and the licensee identified one issue (respective PERs are listed in the attachment) with locked valves. The licensee's guidance for locking methodology is identified in OPDP-6, Locked Valve/Breaker Program, which states that the locking device should be fastened in a manner that resists movement. The issues varied from inadequately locked to not locked, contrary to the licensee's guidance. While the licensee initiated corrective actions to verify that valves outside of radiological dose intensive areas are adequately locked, as noted in 0-PI-OPS-17.0, 18-Month Locked Valve Verification, the licensee did not note that the issues constituted an apparent trend in order to initiate a trend PER or denote an existing PER as a trend PER to capture all of the appropriate corrective actions. Through the remaining inspection period, an additional three issues were identified by the inspectors and one issue by the licensee. The licensee's response to the inspectors regarding trending documentation was that the associated group had not yet completed its quarterly trend assessment. The licensee subsequently initiated trend PER 74361 on January 5, 2004. The licensee used business practice procedure, BP-250, Corrective Action Handbook, to complement their administrative procedure, SPP-3.1, Corrective Action Program. The charter for the licensee's management review committee resides in BP-250 which requires in part "consideration of trends or recurring conditions." The inspectors determined that this is the only guidance in BP-250 regarding trending, other than the integrated assessments done on at least a semiannual basis. The licensee is evaluating changes to their handbook regarding guidance for trending of obvious issues as opposed to more subtle issues identified by a rigorous examination during the integrated assessment.

.3 Annual Sample: Review of Siemens 6.9-kV Breaker Problems

a. Inspection Scope

The inspectors reviewed licensee actions to resolve problems with Siemens breakers. This review began as a look at how the licensee addressed problems associated with the replacement of ABB breakers with Siemens breakers because the licensee had initiated numerous PERs since October 2001, and because the breakers had been installed in locations where common problems could affect multiple safety-related systems. However, due to a series of events where different breakers failed to close during testing and one instance of a Siemens breaker for RHR Pump 1A failing to close on demand, the inspectors focused the review on the causes of the failures themselves and corrective actions for previously identified problems.

In November 2001, the licensee began replacing the existing air circuit breakers in the safety-related 6.9-kV shutdown boards, supplied by ABB, with vacuum circuit breakers from Siemens. These breakers were an already-marketed design, but modified to fit the existing ABB cubicles and qualified by Wyle Labs using a commercial grade dedication process. Since November 2001, the licensee has initiated approximately 50 PERs concerning problems with

Siemens breakers, at least three of which were deemed significant. In two of these, PER 18572 and PER 21862, the licensee rolled several problems into one. The third, PER 60199, also a rollup PER, was written to address the following breaker failures:

On January 31, 2003, a Siemens breaker failed to close while racked to the test position during initial checks after installation.

On June 6, 2003, the Siemens breaker for ERCW Pump M-B failed to close while racked to the test position during post-maintenance testing.

On July 31, 2003, a Siemens breaker failed to close while racked to the connect position during post-maintenance testing.

On February 11, 2004, the Siemens breaker for ERCW Pump P-B failed to close while racked to the connect position during post-maintenance testing.

On February 18, 2004, the Siemens breaker for Containment Spray Pump 2A failed to close while racked to the connect position during post-maintenance testing.

On April 9, 2004, the Siemens breaker for ERCW Pump M-B failed to close while racked to the connect position during post-maintenance testing.

On April 26, 2004, the Siemens breaker for ERCW Pump P-B failed to close while racked to the connect position during post-maintenance testing.

On July 7, 2004, RHR Pump 1A, which used a Siemens breaker, failed to start on demand during surveillance testing. This was the first in-service demand failure of a Siemens breaker.

b. Findings and Observations

Introduction: The inspectors identified an apparent violation (AV) for failure to promptly identify and correct binding problems with the Siemens breaker mechanism operated cell (MOC) slide assembly that resulted in the failure of RHR Pump 1A. This has potential safety significance greater than very low safety significance and will remain unresolved pending completion of the SDP.

Description: On July 7, 2004, RHR Pump 1A failed to start during routine surveillance testing because the breaker did not close and latch. The licensee immediately declared the pump inoperable and began troubleshooting. The same failure occurred a second time during troubleshooting. At that point the licensee replaced the Siemens breaker with an older style ABB breaker and declared the pump operable after end device testing on July 8, 2004. Later, the licensee, along with vendor personnel, examined the failed breaker, determined that the MOC slide assembly was binding on the mounting hardware, and attributed the failure to insufficient clearance between the assembly and the mounting hardware. They also indicated that this binding was exacerbated by bradding of the slide assembly metal at the upper end of the mounting slot that allowed the slide assembly to become wedged between the circuit breaker side sheet and mounting hardware. The bradding was caused by the successive impacts of the slot against the mounting hardware as the breaker was cycled open. The failed breaker had been installed in the RHR Pump 1A cubicle on April 27, 2001, and was last successfully operated on June 23, 2004.

The inspectors reviewed the PER descriptions of previous problems with Siemens breakers, observed the licensee examination of the failed RHR breaker, examined the MOC slide assembly on the failed breaker, and interviewed the involved engineering and maintenance personnel. In addition, the inspectors compared digital photographs of the MOC slide assembly from the failed RHR breaker against those of a MOC slide assembly from a different breaker that failed during testing at the vendor facility. From these actions, and after reviewing the circumstances surrounding the breaker failure, the inspectors concluded that the licensee had several previous opportunities to identify and correct the problem with the RHR breaker before the failure occurred.

The vendor had made five revisions to the basic breaker design due to problems that occurred at the site. Four of these revisions involved problems with the mechanism for driving the MOC switch. The inspectors concluded that these design changes provided an opportunity for the licensee to do a broad, thorough review of the MOC design and, therefore, offered an early chance to see the potential for binding between the MOC slide and the mounting hardware.

Following the failure of the ERCW P-B Pump on April 26, 2004, the licensee sent that breaker and three others to a Siemens facility for root cause evaluation. On May 3, 2004, the licensee received a draft report from Siemens on the root cause of the failures of these breakers. This draft report indicated that one of the breakers had failed because the MOC slide assembly became stuck in the open position due to bradding caused by the impact of the assembly mounting slot against the mounting hardware as the breaker opened. The vendor recommended an inspection of all deliverable breakers to ensure that excessive bradding had not occurred. They suggested that this inspection could be visual or functional, but stated that a visual inspection was somewhat subjective and recommended that guidance for evaluation of the bradding be done by the licensee representative who witnessed the earlier testing at the vendor facility. While allowing that some minor bradding was normal, the vendor suggested that a functional test, which included disconnecting the MOC actuator at its gear drive and exercising it to prove that no binding occurs, was a less subjective and more accurate method of inspection.

The licensee elected to do visual inspections, not the functional tests. On May 4, 2004, the licensee performed visually inspected 12 breakers designated as spare and not installed in the plant. The inspection was performed by the licensee breaker specialist who had observed the root cause testing at the Siemens facility. Of the 12 breakers tested, three were considered to have slight bradding with the remaining nine considered to have no bradding. On May 28, 2004, the licensee initiated a visual inspection of six breakers installed in the A train of the emergency core cooling systems (ECCS), including RHR Pump 1A, which was inspected on June 9, 2004, and 12 breakers installed in the B train of ECCS. Because the vendor indicated that some bradding was normal and small amounts of bradding had been found on earlier inspections, the licensee chose to perform these inspections with electrical maintenance personnel using a boroscope. Each breaker was in its cubicle and connected to the bus. The inspection was recorded and engineering personnel were to determine the acceptability of the inspection data. However, engineering personnel were not present at any of the examinations of the 18 installed breakers and reviewed the video tape on only one that was questioned by the technicians, which engineering determined to be grease. The other 17 breakers were considered to have no bradding. The inspectors determined that the binding problem was actually present at the time of this inspection but was missed because the licensee chose to use the more subjective visual test instead of the functional test.

In order to verify that licensee actions had been sufficient to promptly identify and correct the problem with the MOC slide assembly, the inspectors reviewed PERs written on Siemens breakers to determine whether or not there had been any previous occurrences of similar binding problems with other breakers. This search revealed that on July 11, 2003, while doing receipt inspection of a Siemens breaker, maintenance personnel identified a problem with the MOC slide assembly not being able to move freely in the elongated slot. At that time the licensee loosened the shoulder bolt holding the MOC slide assembly for the affected breaker and entered the problem into PER 26065. The inspectors concluded that this problem, if not identical to, was at least a precursor to the binding problems seen on the RHR pump breaker.

From this information, the inspectors determined that the problem that led to the failure of RHR Pump 1A on July 7, 2004, was actually present when the licensee performed visual inspections on June 9, 2004, but was missed. Also a similar binding problem on a different breaker had been discovered on July 11, 2003. The inspectors concluded that the licensee's actions upon discovery of binding in the breaker at the Siemens facility in April of 2004 did not improve the possibility of identifying and correcting the problem with the RHR breaker. The licensee chose not to perform a thorough search for previous occurrences of similar binding problems and a functional inspection of the breakers for bradding. Because the binding problem existed at the time of inspection on the RHR breaker, a similar binding problem had occurred earlier on a different breaker, and the licensee chose not to perform the more rigorous functional inspection recommended by Siemens, the inspectors concluded that the licensee failed to identify and correct a known problem that resulted in the failure of RHR Pump 1A to start on demand.

<u>Analysis:</u> This finding was considered more than minor because, given that Siemens breakers were used in both trains of several ECCS subsystems, the failure to identify and correct a problem that resulted in a pump failure to start on demand could reasonably be viewed as a precursor to a significant event. This finding was also determined to potentially have greater significance because the loss of one train of RHR would result in reduced sump recirculation capability following a small or medium break size loss-of-coolant accident and no recirculation capability following the loss of 125-VDC Battery Board 2.

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, requires that measures be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this, on June 9, 2004, the licensee failed to identify and correct a problem with binding on the MOC slide assembly of the breaker for RHR Pump 1A that subsequently resulted in the failure of that pump to start on demand. Pending determination of safety significance, this finding is identified as an apparent violation (AV) 05000998/2004007-07, Failure to Identify and Correct MOC Binding Problems on Siemens Breakers.

4OA3 Event Followup

.1 Unit SCRAM - July 4

a. Inspection Scope

The inspectors responded to an automatic scram that occurred on July 4. The inspectors discussed the scram with operations, engineering, and licensee management personnel to gain an understanding of the event and assess followup actions. The inspectors reviewed operator actions taken in accordance with licensee procedures and reviewed unit and system indications to verify that actions and system responses were as expected. The inspectors discussed the scram with the licensee's root cause analysis team and assessed the team's actions to gather, review, and assess information leading up to and following the scram. The inspectors later reviewed the initial investigation report and root cause determination to assess the detail of review and adequacy of the root cause and proposed corrective actions prior to unit restart.

The licensee's investigation identified that the root cause of the turbine trip was a loss of turbine speed signal following the turbine/generator response to fault on a 500-kV transmission line. At the end of the inspection period, the licensee was reviewing a previous design change that incorporated a feature to monitor speed probe sensor input such that a failed sensor would not result in a spurious turbine trip. This circuit appeared to have caused the total loss of turbine speed signal. The inspectors also reviewed the initial licensee notification to verify that it met the requirements specified in NUREG-1022, Event Reporting Guidelines. Inspector observations were compared to the requirements specified in the procedure listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Reactor Building Crane Trolley Drop

a. Inspection Scope

The inspectors responded to the Reactor Building Crane Trolley drop that occurred on August 24. The inspectors discussed the event with licensee management, engineering, vendor support, and maintenance personnel to gain an understanding of the conditions leading up to the drop and actions taken immediately following to assess licensee actions. The inspectors reviewed the root cause report to assess the detail and thoroughness of the report and proposed corrective actions. The inspectors also reviewed the event for reportability in accordance with NUREG 1022, Event Reporting Guidelines.

b. Findings

This issue was dispositioned in Section 1R17.

.3 Inadvertent SRV Opening During Testing

a. Inspection Scope

The inspectors observed control room personnel responding to an unexpected opening of an SRV on August 18, 2004 on unit 1. The inspectors arrived in the control room shortly after the SRV was re-closed and observed the followup actions by the licensed operator, including operator briefings, actions required by the off-normal procedures and monitoring of plant conditions. As part of the followup to this event, the inspectors observed plant chart recorders, compared requirements of off-normal procedures to observations of operators' performance, and discussed with plant personnel the various methods available to the operators to close the SRV. The following documents were reviewed and used as criteria for evaluating the operators' response to this event:

- DES-21-1 "SRV Inadvertent Opening/Stuck Open"
- DES 00-3901 "Unanticipated Opening of SRV 1B21F0052D During Surveillance Test"
- DES 00-3903 "SRV Weeping After Being Opened and Closed"

b. Findings

Introduction. A Green self-revealing NCV was identified for failure to have an adequate surveillance procedure in accordance with TS 5.4.1.a., which resulted in the inadvertent opening of an SRV during testing.

Description. On August 18, 2004, a self-revealing finding was identified when SRV 1B21F0051D unexpectedly opened, at 2:15 p.m., during a calibration using Surveillance Instruction (SI) DES-B21-T0369, "SRV Surveillance Calibration." Licensed operators responded to the event by promptly following Procedure DES B21-1, "SRV Inadvertent Opening/Stuck Open," which required reducing power to 90 percent and the closing the SRV. The SRV was closed successfully within 2 minutes of its opening. As expected, there was an increase in the suppression pool temperature and level, although these parameters remained within TS limits.

The licensee's investigation determined the cause to be an inadequacy of SI DES-B21-T0369. The SI did not have a step to reset the low-low set logic before applying an input signal to the trip unit. The licensee also determined that it missed an opportunity to prevent the event during identical testing the previous week. During the previous test, licensed operators and instruments technicians questioned why the low-low set logic lights were lit and evaluated the condition. They decided to reset the logic before continuing with the test. This action was not documented and the procedure weakness was not recognized at the time. When questioned by the inspectors as to why previous uses of the procedure didn't cause the valve to open, the licensee stated that the most recent revision to the procedure left out the specified step.

Analysis. The deficiency associated with this event is an inadequate procedure, which led to the unexpected opening of the SRV at full power during calibration. The finding was greater than minor because it had an actual impact of lifting a relief valve which is a precursor to a significant event (e.g., relief valve stuck open). The finding which is under the initiating events cornerstone was only of very low safety significance because, although the likelihood of a reactor trip increased, all mitigating systems were available. The licensee entered the deficiency with the surveillance procedure into their corrective action (CA) program for resolution. The cause of the finding is related to the cross-cutting element of problem identification and resolution.

Enforcement. TS 5.4.1.a requires written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A. Regulatory Guide 1.33, Appendix A, Item 8b, requires procedures be maintained for the surveillance tests listed in the TS. Contrary to the above, SI DES-B21-T0369 was not maintained, in that its performance on February 18, 2004, resulted in an inadvertent opening of an SRV during testing on unit 1. Because this failure to maintain adequate surveillance instructions is of very low safety significance and has been entered into the CAP (CR 00-3901), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000998/2004007-06 Failure to Maintain Adequate Surveillance Instruction to Prevent Inadvertent SRV Opening.

(Note: Example of closing an LER, below)

.4 (Closed) LER 05000999/2004003-0 1, Inadvertent Engineered Safety Feature Actuation Caused by Loss of RPS Power Supply

On February 4, 2004, Unit 2 “B” RPS power was lost because the associated voltage regulator card failed. The failure resulted in an RPS “B” half scram and corresponding containment isolations. The licensee replaced the voltage regulator card and reestablished the “B” motor-generator set as the normal power source for the “B” RPS system. The LER was reviewed by the inspectors and no findings of significance were identified and no violation of NRC requirements occurred. The licensee documented the failed equipment in CR 269440. This LER is closed. ***(Note: Reference to the CR)***

(Note: Example of closing an LER with a minor violation, below)

.5 (Closed) LER 05000998, 999/2004009-02 Primary Containment Isolation Valves not Checked per Surveillance Requirements

On March 10, 2004, the licensee identified that 87 Unit 1 and 85 Unit 2 primary containment isolation valves (PCIVs) had not been tested as part of monthly TS SR 3.6.1.3.2. The licensee determined that what caused the PCIVs to be excluded from the surveillance was an unclear definition of the components that constitute containment boundary. All of the valves were subsequently tested, with no identified leakage. Additional corrective actions, completed or planned, included revising the associated surveillance procedure and clarifying the wording in the TS bases. No new findings were identified in the inspector's review. This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee documented the problem in CR 276714. This LER is closed.

(Note: Example of closing out an LER with a Licensee-Identified NCV, below)

.6 (Closed) LER 05000999/2004004-04. Technical Specification Interpretation Incorrect ─ Operation Prohibited by TS

On February 17, 2004, the licensee identified that one Unit 2 PCIV was inoperable and the associated TS limiting condition for operation had not been entered. Specifically, on February 11-14, 2004, one of the PCIVs in a hydrogen/oxygen (H_2O_2) analyzer penetration was inoperable, and the penetration was not isolated as required by TS 3.6.1.3. The licensee determined the cause to be unclear wording in the FSAR for the design basis for the H_2O_2 analyzer penetration and a non-conservative Technical Specification Interpretation (TSI) for the associated section. Corrective actions included a revision to the specific TSI, a review of all existing TSIs for non-conservative direction, and a plan to eliminate all TSIs. This finding is more than minor because it had a credible impact on safety, in that if the redundant valve in the penetration did not close on a containment isolation signal, containment integrity would not be ensured. The finding affects the Barrier Integrity Cornerstone and was considered to have very low safety significance (Green) using Appendix H of the SDP because the likelihood of an accident leading to core damage was not affected, the probability of early primary containment failure and therefore a large early release was negligible, and the redundant isolation valve remained operable during this event. This licensee-identified finding involved a violation of TS 3.6.1.3, Primary Containment Integrity. The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

(NOTE: Since Licensee Identified Violation, only significance of LER issue discussed here. Section simply refers to 4OA7 for enforcement)

4OA5 Other (Optional 4-point format)

.1 Institute of Nuclear Power Operations (INPO) Plant Assessment Report Review

a. Inspection Scope

The inspectors reviewed the final report for the INPO plant assessment of Dirojac station conducted in August 2004. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings of significance were identified.

.2 Independent Spent Fuel Storage Installation (ISFSI) Radiological Controls

a. Inspection Scope

The inspectors conducted independent gamma and neutron surveys of the ISFSI facility and compared the results to previous quarterly surveys. The inspectors also observed and evaluated implementation of radiological controls, including RWPs and postings, and discussed the controls with a HPT and HP supervisory staff. Radiological controls for loading Hi-Storm ISFSI casks were also reviewed and discussed.

Radiological control activities for ISFSI areas were evaluated against 10 CFR Part(s) 20 and 50, NRC Certificate of Compliance (COC) #1014, and applicable licensee procedures. Documents reviewed are listed in section 4OA5 of the report attachment.

b. Findings

No findings of significance were identified.

.3 (Closed) NRC Temporary Instruction (TI) 2515/154, Spent Fuel Material Control and Accounting at Nuclear Power Plants

During the previous reporting period, the inspectors completed Phase I and Phase II of Temporary Instruction 2515/154, Spent Fuel Material Control and Accounting at Nuclear Power Plants. Appropriate documentation of the results was provided to NRC management, as required by the TI. This completes the Region X inspection requirements for this TI.

.4 (Closed) NRC TI 2515/156, Offsite Power System Operational Readiness

During the previous reporting period, inspectors collected data from licensee maintenance records, event reports, corrective action documents and procedures, and through interviews of station engineering, maintenance, and operations staff, as required by TI 2515/156. Appropriate documentation of the results was provided to headquarters staff for further analysis, as required by the TI. This completes the Region II inspection requirements for this TI.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 1, 2004, the resident inspectors presented the inspection results to Mr. B. Handcuff and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

Note: This is standard language when inspectors review violations that have been identified by the licensee, have been entered into the corrective action program and which are being handled properly.

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

Note: Must state the requirement; NRC tracking numbers are not required since these violations will not be put into the PIM or RPS.

- TS 3.6.1.3 requires that a primary containment penetration be isolated within 4 hours, if the associated PCIV is not operable. Contrary to this, on February 11 to 14, 2004, a PCIV for a Unit 2 H_2O_2 analyzer was not operable, and the penetration was not isolated within 4 hours. This was identified in the licensee's CAP as CR 272962. This finding is of very low safety significance because it does not represent an open pathway in the physical integrity of the reactor containment.

- 10 CFR 20.1501(a)(1) requires that surveys be made to comply with the regulations in 10 CFR Part 20, including 10 CFR 20.1902(b) for posting of high radiation areas (defined as an area greater than 100 mr/hr at 30 centimeters). On March 12, 2004, a shipping cask had not been surveyed properly and, as a result, an area measuring 700 mr/hr at 30 centimeters was undetected and constituted a high radiation area that was not posted. This event is documented in the licensee's CAP as CR 297422. This finding is only of very low safety significance because it did not involve a very high radiation area or personnel over-exposure.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

(Note: This Supplementary Information is an attachment to the report [which is an enclosure to the cover letter], and will be numbered starting with page A-1. Each page should have "Attachment" placed as a footer flush to the right)

(Note: This list is for illustration. It does not reflect the actual inspection report)

KEY POINTS OF CONTACT

Licensee personnel

S. Lee, Vice President - Site Operations
R. Shawin, Vice President - Support
B. Mills, Station Manager
K. Hicks, General Manager
B. Harris, Manager, Training
K. Leach , General Manager - Assurance
S. Vissing, General Manager Nuclear Licensing
A. Roe, Radiation Protection Superintendent

NRC personnel

A. Brown, Resident Inspector (Trainee), Reactor Projects Branch B

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000998/2004007-07	AV	Failure to Identify and Correct MOC Binding Problems on Siemens Breakers (Section 4OA2.2).

Opened and Closed

05000998/2004007-01	NCV	Failure to Provide Complete and Accurate Information for One Licensed Operator (Section 1R11)
05000998, 999/2004007-02		NCV Failure to Demonstrate Performance of the Reactor Motor-Operated Valve (RMOV) Board 1B Through Preventive Maintenance (Section 1R12)
05000999/2004007-03	NCV	Failure to Comply with TS 3.3.1 to Trip RPS Bistables (Section 1R15)
05000998/2004007-04	NCV	Failure to Comply with 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures and Drawings (Section 1R17)
05000998/2004007-05	NCV	Inadequate Surveillance Procedure for Remote Shutdown System Instrumentation (Section 1R22)

05000998/2004007-06	NCV	Failure to Maintain Adequate Surveillance Procedure to Prevent Inadvertent SRV Opening (Section 4OA3.3)
05000998/2004007-08	NCV	Exposure to Unanticipated Radiation Levels (Section 2OS1)

Closed

05000998, 999/2515/154	TI	Spent Fuel Material Control and Accounting at Nuclear Power Plants (Section 4OA5.3)
05000998,999/2515/156	TI	Offsite Power System Operational Readiness (Section 4OA5.4)
05000999/2004003-01	LER	Inadvertent Engineered Safety Feature Actuation Caused by Loss of RPS Power Supply (Section 4OA3.4)
05000998, 999/2004009-02	LER	Primary Containment Isolation Valves not Checked per Surveillance Requirements (Section 4OA3.5)
05000999/2004004-04	LER	Technical Specification Interpretation Incorrect — Operation Prohibited by TS (Section 4OA3.6)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

(Note: Typical reference list and an example of what it should look like for some sections. May also have design changes and other type procedures and documents. Documents listed should be those used to decide licensee performance in applicable sections, rather than all documents reviewed. State the revision number of the document if available. Documents in this list do not exactly match the associated sections in the sample report)

<u>Section 1R01: Adverse Weather Protection</u>

OP 11887-1(2), Cold Weather Checklist
OP 11901-1(2), Heat Tracing System Alignment
OP 13901-1(2), Heat Tracing System
OP 17104-1(2), Annunciator Response Procedures for Heat Tracing Panels
    25743-C, Thermon Solid State Heat Tracing and Freeze Protection System Calibration and Maintenance
OP 18887-1(2), Condensate System Checklist
OP 19901-1(2), Emergency Service Water System Alignment

<u>Section 1R04: Equipment Alignment</u>

2-SO-63-5, Emergency Core Cooling System, Revision 35
0-SO-65-1, Emergency Gas Treatment System Air Cleanup and Annulus Vacuum, Revision 13
1,2-47W810-1, Residual Heat Removal System Flow Diagram, Revision 43

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

1,2-47W494-4, Fire Protection - Plan Elevation 734.0, Revision 7
SPP-10-10, Control of Transient Combustibles, Revision 3S1

<u>Section 1R06: Flood Protection Measures</u>

UFSAR Sections 2.3 and 2.4, including Appendix 2.4A, Flood Protection Plan
PER 24226, Switches 0-LS-18-3 and 0-LS-18-6 Under Water due to High Level in the FOST Moat
PER 24739, CCW Building Penetration for the Old Fire Header Strainer Discharge not Sealed
PER 33672, Flood Mode Spool Piece 1-SPPC-067-0687 Did not Fit to Valves 1-70-662 or 1-67-678
PER 61940, While Releasing Clearance, Drain Valves for Fire Protection Deluge Valve Left Open
PER 62252, Leak Determined to be Present on the HPFP System
PER 63385, Two “Turb Aux or Reac Bldg Flooded” Alarms Received Five Minutes Apart
PER 65647, Scheduled Maintenance Activity for the HPFP System Removes Numerous Hose Stations and Sprinkler systems From Service
PER 65838, Leak in HPFP System
PER 66671, Fire Pump Start Signal on the Main Fire Protection Console

Calculation SQS40056, Moderate Energy Line Break Flooding Study, Revision 10
Letter from R C Williams to J H Rinne, Dirojac Nuclear Plant Cable Splices in Underground Ductbanks, dated July 26, 2000
PER 22700, ERCW Pump P-B Tripped on Overcurrent
WO 03-018293-000, Check Standing Water in Manholes/Handholes
10N213, Grading Plan - Intake Channel, Revision 9
17W304-1, ERCW Supply Piping, Revision 13
17W304-2, ERCW Supply Piping, Revision 9
17W304-3, ERCW Supply Piping, Revision 5
17W304-4, ERCW Supply Piping, Revision 5
17W304-5, ERCW Supply Piping, Revision 5
1,2-47W845-2, Mechanical Flow Diagram-Essential Raw Cooling Water System, Revision 82
1,2-47W859-1, Mechanical Flow Diagram-Component Cooling System, Revision 49
1-47W859-2, Mechanical Flow Diagram-Component Cooling System, Revision 30
2-47W859-3, Mechanical Flow Diagram-Component Cooling System, Revision 30
1,2-47W850-2, Mechanical Flow Diagram-Fire Protection, Revision 26
1,2-47W850-24, Mechanical Flow Diagram-Fire Protection, Revision 20
1,2-47W803-2, Mechanical Flow Diagram-Auxiliary Feedwater, Revision 59
1,2-47W809-7, Mechanical Flow Diagram-Flood Mode Boration Makeup System, Revision 20
AOP-N.03, Flooding, Revision 21
0-SO-84-1, Flood Mode Boration Makeup System, Revision 7
0-PI-FPU-026-073.A, Fire/Flood Mode Pump A-A Flow Test, Revision 0
0-PI-FPU-026-073.B, Fire/Flood Mode Pump B-B Flow Test, Revision 1
1-PI-SFT-084-001.0, Functional Test of Flood Mode Boration Makeup System, Revision 5
2-PI-SFT-084-001.0, Functional Test of Flood Mode Boration Makeup System, Revision 6

Section 1R07: Heat Sink Performance

DIR-VTD-D270-0130, Dunham-Bush PCX Package Chillers

Section 1R11: Licensed Operator Requalification Program

Nuclear assurance - audit report no. SSA0305 – Dirojac Self- Assessment Report SA-TRN-03-002
Dirojac Nuclear Plants and corporate (coc) - DIrojacN-wide - operations
Functional area audit (including nuclear fuels and reactor engineering)
Scenarios 3-OT-SRE 022 Large Break LOCA, 3-OT-SRE-007 SGTR with loss of 6.9KV SDB
Badge Access Transaction Reports for Reactivation of Licenses (3)
Licensed Operator Medical Records (12)
Remedial Training Records:

- Inspectors reviewed two remedial training records, one for a written exam failure, and one for a JPM exam failure.

Written Exams Reviewed:

- RO 2004/2003 Exams, # 4, 5, and 7
- SRO 2004/2003 Exams, # 5, and 7

Simulator Fidelity Documents:

- Malfunction Tests:
  IA02 “Loss of Non-Essential Control Air.”
  RD07 “Dropped Rod.”
  ED15 “Loss of 250VDC Battery Board.”
  FW05/06/07/22 “Loss of All Feedwater.”
- Transient Tests:
  Transient Test # 9 “Maximum Size Main Steam Line Break, TT-9.”
  Transient Test # 4 “Simultaneous Four Loop Reactor Coolant Pump Trip, TT-4.”
  Transient Test # 10 “Primary System Depressurization Using PZR Relief Valve, TT-10.”

Simulator Problem Reports Reviewed:

- 2561 CERPI Indication On A Dropped Rod.
- 2532 CERPI Shows Rod Demand Speed In Manual.
- 2498 Adjust PRT Pressure To More Closely Match the Plant.
- 2486 Investigate RCS Temperature Change in TT1.

Section 1R12: Maintenance Effectiveness

1-SI-IFT-068-322.4 Functional Test of Pressurizer Pressure Channel IV, Rack 13, Loop P-68-322, Revision 7
WO 04-779355-000, Repair/Adjust Containment Spray Pump 2B-B Shelf Switch

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Online Sentinal Run for July 6, 2004, through July 23, 2004
Online Sentinal Run for July 26, 2004, through August 12 , 2004
Online Sentinal Run for August 23, 2004, through August 27, 2004
0-TI-DSM-000-007.1, DIrojac Risk Assessment Guidelines, Revision 8
SPP-7.1, DIrojacN On-line Work Management, Revision 5
Online Sentinal Run for September 13, 2004 through September 20, 2004

Section 1R15: Operability Evaluations

PER 64454, Air Flow Bypass of Auxiliary Building Room Coolers with Access Panels Removed
AOP-I-11, Eagle 21 Malfunction, Revision 5
AOP-I.02, RCS Loop RTD Instrument Malfunction, Revision 1
AOP-I.04, Pressurizer Instrument Malfunction, Revision 6
2-2000E54-1,2, and 3, Revision 1, Rack 13 Protection Set IV Wiring Diagrams
Dirojac White Paper, Reactor Protection System Soft Trip vs. Hard Trip
PER 62486, 2B-B Emergency Diesel Generator Damper Discovered Shut
WO 04-772018-000, Repair/Replace the Northwest Intake Damper Actuator on the 2B-B Diesel Generator
UFSAR Section 9.4.5, Diesel Generator Building
1,2-47W866-9, Heating Ventilating Air Flow Diagram for Diesel Generator Building
PER 61789, Upper Detector Ammeter Channel N41 Out of Tolerance
1-SI-IFT-092-N41.1, Functional Test of Power Range Nuclear Instrumentation System, Channel N41

FSAR Section 7.2, Reactor Trip System

Section 1R16: Operator Work-Arounds

Operations Directive Manual - 3.7, “Operator Work-Around Program,” Revision 8
Dirojac Select Focus Area Report, dated August 27, 2004
ARD 1, Unit 1 Auxiliary Building
ARD 2, Unit 2 Auxiliary Building
ARD 3, Unit 1 Turbine Building
ARD 4, Unit 2 Turbine Building
ARD 5, Control Building
ARD 6, Radwaste
ARD 7, Outside
ARD 8, Con DI

Section 1R17: Permanent Plant Modifications

N3-82-4002, Standby Diesel Generator System description
N3-30DB-4002, Diesel Generator Building Ventilation System description
WO 03-011112-000, Implement DCN 51383-A Stage 1 for DG Fan 1A-A temp switches
PER 71968, Licensee identified problem of DG exhaust fans not auto-starting during performance of the 1B-B DG start and load test.
TI-215, Work Permits, Appendix C, Painting, Cleaning, Sealing and Other Volatile Hydrocarbon Use Permit
WO 04-810947-000, Implement DCN to apply proprietary coating to the Unit 1 fuel transfer canal to eliminate leakage
Test report for high temperature testing of seven specimens of blue polyurea material. Schenectady Material and Processes Laboratory, Inc., Lab No. KR-0407 for Purchase Order No. DS-498, dated March 31, 2000

Section 1R19: Post--Maintenance Testing

0-PI-SFT-067-002.0, ERCW Pump Power Draw Measurement, Revision 2
0-SI-SXP-067-201.R, Essential Raw Cooling Water Pump R-A Performance Test, Revision 4
0-SI-SXP-067-201.Q, Essential Raw Cooling Water Pump Q-A Performance Test, Revision 6
1,2-45N767-1, 6900V Diesel Generator Schematic Diagrams Sheet 1, Revision 26
1,2-45N767-3, 6900V Diesel Generator Schematic Diagrams Sheet 3, Revision 24
WO 04-779355-000, Repair/Adjust Containment Spray Pump 2B-B Shelf Switch
0-SI-SXV-074-266.0, ASME Section XI Valve Testing - 1B RHR Mini-Flow Valve

Section 1R20: Refueling and Other Outage Activities

Procedures

RF 29542-C, Shutdown Risk Management
RF 29540-C, Risk Assessment Monitoring
RF 12005-C, Reactor Shutdown to Hot Standby

RF 12007-C, Refueling Operations
RF 18019-C, Loss of Residual Heat Removal
RF 18030-C, Loss of Spent Fuel Pool Level or Cooling
RF 13005-1, Refueling Cavity Draining
RF 11899-1, Draindown Configuration Checklist
RF 14406-1, Boron Injection Flow Path Verification - Shutdown
RF 93641-C, Development and Implementation of the Fuel Shuffle Sequence Plan
RF 93663-C, Verification of Core Loading Pattern
RF 14210-1, Containment Building Penetrations Verification - Refueling
FME 00254-C, Foreign Material Exclusion and Plant Housekeeping Programs
RFO Schedule - 2R10, Revision 4

Section 1R22: Surveillance Testing

1-SI-0-4, Appendix C, Page 7 of 11 completed on the following dates: 9/4/04, 8/7/04, 7/9/04, 6/12/04, 5/14/04, 4/17/04, 3/20/04, 2/21/04, ½4/04, 12/27/03, 11/28/03, 11/1/03, 10/17/03
SSD-1-LPT-68-65C-S, Rev. 2, Scaling and Setpoint Document for RCS Loop 4 Hot Leg Temp
PER 68838, NRC identified that Loop 4 hot leg temp indicator was outside of its MCD resulting in an unplanned entry into LCO 3.3.4 Action A.
PER 70638, NRC identified that Loop 4 hot leg temp indicator was outside of its MCD resulting in an unplanned entry into LCO 3.3.4 Action A.
WO 04-822471-000, repair 1-LPT-68-0065C, Loop 4 hot leg temp indicator outside of its MCD (calibrations performed)
WO 04-822570-000, repair 1-LPT-68-0065C, Loop 4 hot leg temp indicator outside of its MCD (modifier replaced)
SPP-2.2, Administration of Site Technical Procedures
PER 72202, NRC identified problem regarding procedure 1-SI-3-901-B steps signed N/A contrary to requirements
PER 71291, NRC identified the steps in continuous use procedure 1-SI-30-9-B were not being signed off when completed
Instrument Maintenance Instruction (IMI) - 99.060, Transmitter Bench Response Time Test

Section 2OS1: Access Control To Radiologically Significant Areas

Radiation Work Permits

2004-1620	Perform walkdowns/take field measurements in main steam tunnel for permanent shielding design
2004-1800	RFO-12 refueling activities
2004-1912	RFO-12 remove/replace 16 SRVs
2004-1915	RFO-12 remove/replace LPRMs, including all support activities
2004-1933	RFO-12 ISI weld inspections in drywell
2004-1935	RFO-12 drywell valve maintenance
2004-1936	RFO-12 installation/removal of temporary shielding in the drywell
2004-1952	Perform walkdowns/take field measurements in drywell for permanent shielding design

2004-1953 RFO-12 ISI welds inside bioshield on N2 nozzels, including support activities

<u>Procedures</u>
RP-105 Radiation Work Permits, Revision 4
RP-108 Radiation Protection Postings, Revision 2
RP-204 Special Monitoring Requirements, Revision 3
RP-501 Respiratory Protection Program, Revision 0
RPP-0005 Management of Radiological Postings, Revision 25
RPP-0006 Performance of Radiological Surveys, Revision 18
RSP-0212 Drywell Entry, Revision 10A
ADM-0071 Fuel Pools Material Control, Revision 4

<u>Audits and Assessments.</u>
QS-2003-RBS-009
QS-2003-ENS-017
QS-2004-RBS-005

<u>Corrective Actions Program Documents</u>
PER - 18255, Failure to notify SM [Shift Manager] of removal of LHRA postings as required in RCI-29.
PER - 26782, Several discrepancies, inconsistencies and improvement areas in radiological postings, tags and barriers were identified by the Nuclear Assurance Audit Team reviewing the Radiological Control Program.
PER - 27503, Two Operations Individuals (ID numbers omitted) received unanticipated Dose Rate alarms on their Electronic Dosimeter when they entered an area other than what they had informed RADCON.
PER - 64828, A previous PER identified the need for posting of survey maps of the work area for the Dry Cask Work. Upon receipt of a survey, it was evident that the general area had neutron and gamma dose rates that should be avoided or mitigated.

<u>Section 2OS3: Radiation Monitoring Instrumentation and Protective Equipment</u>

<u>Procedures</u>
RCI-04, Respiratory Protection Program, Revision 44
RCI-05, Radiological Control Instrumentation Program, Revision 39
0-PI-FPU-049-401.M, Self-Contained Breathing Apparatus, Revision 18
HPT063.002, SCBA Training, Revision 7
SPP-3.1, Corrective Action Program, Revision 7S1

<u>Records</u>
Waste Package Area ARM 90-3, Calibrations, 06/16/99 and 05/08/01
Containment Post-Accident Hi Range ARM Nos. 2-R-90-271, 2-R-90-272, 2-R-90-273, 2-R-90-274, Calibrations, April 2004 and November 2003
AMS-4 No. 1603, Calibrations, 01/26/04 and 06/22/04
10 CFR Part 61 Analysis, Dry Active Waste, 05/08/03
SCBA Breathing Air Quality Analysis, 07/08/04

Air Cylinder Nos. 45-40 and 45-43, Hydrostatic Testing History, August 1999 - August 2004
SCBA Unit Nos. 45-4 and 45-51, Maintenance History, August 1999 - August 2004
Respiratory Qualification Records, 12 Operations and 3 Fire Brigade Personnel, Randomly
Chosen.

Corrective Action Program Documents
Self-Assessment No. SQN-RP-03-003, Respiratory Protection Program, 08/25/03 - 08/29/03
PER - 63987, Internal check sources for some abandoned ARMs not properly inventoried,
06/29/04
PER - 66203, Breathing air cylinders have wrong valve thread, 07/30/04
PER - 66496, Licensed operator did not have corrective lenses available for SCBA use, 08/04/04

Section 2PS1: Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

Procedures
0-TI-CEM-260-049.3, Gamma Spectroscopy Sample Changing System Operation, Revision 1
0-TI-CEM-260-049.1, Gamma Spectroscopy Systems Periodic Performance Checks, Revision 0
0-TI-CEM-260-049.2, Liquid Scintillation System Calibration Check, Revision 1

Surveillances
0-SI-CEM-040-421.0, Turbine Building Sump Discharge Radioactivity Determination and TBS or ERCW Inoperable Radiation Monitors, Rev.9 (Including calibration work performed on 5/3/04 and 5/23/04)
2-SI-ICC-090-400.0, Calibration of Shield Building Radiation Monitor 2-RM-90-400, Revision 3 (Includes documentation for calibration performed 2/14/03)
2-SI-ICC-090-400.0, Calibration of Shield Building Radiation Monitor 2-RM-90-400, Revision 4 (Includes documentation for calibration performed 6/1/04)
SI-401, Steam Generator Blowdown Continuous Release, Revision 27 (Includes documentation for release permits generated on 4/22/04 and 6/4/04)
0-SI-CEM-030-410.2, Containment Upper and Lower Compartment Purge Sampling, Revision 17 (Includes documentation for release permit generated 7/30/04 and 8/3/04)
0-SI-ICC-090-101.B, Calibration of Auxiliary Building Gaseous Radiation Monitor 0-R-090-101B, Revision 6 (Includes documentation for calibration performed 8/2/02 and 8/24/04)
0-SI-ICC-090-122.0, Channel Calibration of Waste Disposal System Liquid Effluent Radiation Monitor 0-R-90-122, Revision16 (Includes documentation for calibration performed 12/20/01)
0-SI-ICC-090-122.0, Channel Calibration of Waste Disposal System Liquid Effluent Radiation Monitor 0-R-90-122, Revision 19 (Includes documentation for calibration performed 5/14/03)
0-SI-CEM-077-400.1, Liquid Waste Effluent Batch Release, Revision 16 (Release Permit data)

Dirojac Nuclear Plant - Annual Radioactive Effluent Release Reports for 2004 and 2003
Dirojac Nuclear Plant - Offsite Dose Calculation Manual, Revision 47
Dirojac/Analytics Cross-Comparison Report, 3d quarter 2003, 4th quarter 2003, 1st quarter 2004
HPGe Efficiency Calibration Certificate, SQN Detector #3, 6/11/02
Analytics Certificate of Calibration - Standard Radionuclide Source 60943 - 3/1/01
Analytics Certificate of Calibration - Standard Radionuclide Source 63574-160, 4/19/02
Detector Control Charts for HPGe and Liquid Scintillation Detectors covering May-August 2004

Corrective Action Program Documents
PER - 21376, During the performance of SI-244 (Periodic Functional Test of Radioactive Effluent Monitoring Instruments), 0-FI-77-42 and 0-FR-77-42 (Waste Condensate Flow) were found out of tolerance.
PER - 22390, 0-RM-90-101 came into alarm with an Instrument Malfunction. The monitor was blocked and the appropriate ODCM entered.
PER - 24617, During the performance of 0-SI-IFT-090-212.0 (Functional Test of Station Sump Discharge Effluent Radiation Monitor) the Trip 2 function was found at 7.40E+4 which is incorrect.
PER - 26052, A Maintenance Rule functional failure of 2-RM-90-400A, Shield Building Exhaust low range noble gas detector, occurred on September 9, 2003, due to a failed 120-VAC to 24-VDC power supply.
PER - 31624, During performance of 0-SI-IFT-090-140.0 (Functional Test of Essential Raw Cooling Water Effluent Liquid Radiation Monitor), the rotameter was found to have a clamshell lodged in the tube causing the float to be stuck.
PER - 34195, Liquid effluent radwaste discharge radiation monitor, 0-RM-90-122 has recently been exceeding the high radiation setpoint and stopping the discharge during Cask Decon Collector Tank (CDCT) releases due to interaction with radwaste system contamination
PER - 60955, While Operations were pumping down the Turbine building sump to a lower level than normal, 0-RM-90-212 indicated a low flow condition. The flow switch was cleaned on a special performance of 0-SI-IFT-090-212.0 (Station Sump Discharge Effluent Monitor) and we found what appeared to be algae on the flow element.
PER - 66519, A statement in Dirojac's Annual Effluent Report for AVERAGE ENERGY refers to Dirojac's ODCM limiting the dose rates for noble gas there, the average energies (E) for gaseous effluents as described in Regulatory Guide 1.21 are not applicable. The basis for this statement needs to be evaluated.

Section: 2PS3 Radiological Environmental Monitoring Program

Reports, Procedures, Instructions, Lesson Plans and Manuals
Dirojac Nuclear Plant - Offsite Dose Calculation Manual, Revision 47
Dirojac Nuclear Plant - Annual Radiological Environmental Operating Report - 2004
Dirojac Nuclear Plant - Annual Radiological Environmental Operating Report - 2003
Radiological Control Instruction, RCI-1, Radiological Control Program, Revision 62
RCI-05, Radiological Control Instrumentation Program, Revision 39
Environmental Radiological Monitoring Program (EMSTD-01), Revision 21
Collection Of Environmental Monitoring Samples SC-01, Revision 18

Callibration Procedure for Radiological Environmental Monitoring Air Sampler System Gas Meter SC-03, Revision 4
Dirojac Nuclear Plant Environmental Data Station Manual, Revision 2
DIrojacN Emergency Preparedness Field Support Servicing of Meteorology Equipment at Environmental Data Stations (EPFS-3) Revision 10
DIrojacN Emergency Preparedness Field Support Environmental Data Station Meteorological Sensor Exchange - EPFS-4, Revision 12
DIrojacN Emergency Preparedness Field Support Calibration of Environmental Data Station Data Logger and Sonic Channels- EPFS-6, Revision 10
DIrojacN Standard Programs and Processes Meteorological Monitoring Program SPP-5.12, Revision 0

<u>Plant Records</u>
PM-7 Nos. 252, 254, 255, Calibrations, 01/15/04 and 06/29/04
PCM-1B No. 576450, Calibrations, 01/21/04 and 07/20/04
GTM No. 860182, Calibrations, 10/4/03, 03/08/04, and 06/10/04
10 CFR Part 61 Analysis, Dry Active Waste, 05/08/03

<u>Corrective Action Program Documents</u>
PER - 20568, Unexpected Entry into LCO 3.3..3.4 ICS Met Tower out of Service
PER - 21680, Met Tower Lightning Strike
PER - 22745, Met Tower data display unreliable
PER - 25945, Incorrect Rainfall calculation
PER - 26656, Indication of 'Bad Met Data'
PER - 33529, Met Tower stopped updating data
PER - 1454, Trouble with air sampling pump
PER - 1207, Questionable air temperature readings
PER - 66581, 5000 DPM check source did not alarm GTM when four people were standing in close proximity to detector, 08/05/04

<u>Audits and Self-Assessments</u>
Radiological Protection and Control Audit Audit Report NO. SSA 0302 dated 12/31/03
Self-Assessment No. CRP-ERMI-01-004, Environmental Radiological Monitoring and Instrumentation

<u>Section 4OA1: Performance Indicator Verification</u>

<u>LERs</u>
LER 050000327/2003001, Manual Reactor Trip as a Result of Main Generator Trip and Loss of Load
LER 050000327/2004001, Automatic Reactor Trip From Inadvertent Relay Operation on a Main Transformer

<u>Procedures</u>
SPP-3.4, Performance Indicator for NRC Reactor Oversight Process, Revision 0, 04/03/2004
SNP Desktop Guideline for Identification and Reporting of NEI 99-02 Performance Indicators
for Occupational Exposure Control Effectiveness
Common Technical Instruction

Chemistry (0-TI-CEM)-000-001.3, Primary Chemistry Specifications, Revision 16

Plant Records
Individual RCA exit doses exceeding 100 mrem between 10/01/2003 and 04/16/2004
2004 Annual Radioactive Effluent Release Report
Monthly 10 CFR 50, Appendix I, Dose Calculations for Liquid and Gaseous Effluents for the
Months of October 2003 through March 2004

Corrective Action Program Documents
PER - 02-013539-000, Individual Entered RCA Without TLD badge, 10/27/2003
PER - 02-014509-000, Emergent Activities Are Not Being Reviewed and Appropriately Reported/Communicated to the RADCON Staff, 11/26/2003
PER - 03-001633-000, Valid ED Dose Alarms and Dose Rate Alarms Not Being Reported via PER Initiation, 02/18/2003
PER - 02-013073, Effluent Monitor 0-RM-90-134/141 Inoperable, 10/11/02
PER - 02-013472, High Radiation Alarm on Effluent Monitor 0-RM-90-212, 10/25/03
PER - 02-014224, Increase in Gaseous Effluent during October 2003 due to Unit 2 Fuel Leak,
1/19/02
PER - 02-015201, Instrument Malfunction on Monitor 1-RM-90-120/121, 12/17/02
PER - 03-002082, Incorrect Value for Instrument Background Count on Effluent Monitor 0-RM-90-122 Used in Liquid Effluent Batch Release Permit, 03/04/03

Section 4OA2: Identification and Resolution of Problems

SPP-3.1, Corrective Action Program, Revision 6
PER 04-000556-000, 2A CS Pump Failed to Start During Section XI Test
PER 04-00750-000, Perform an Extent of Condition to Determine if End Device Testing Has Been Waived
PER 04-000475-000, ERCW Pump P-B Breaker Failure to Close for PMT
PER 01-009568-000, Consolidation of Siemens Breaker Issues
PER 03-008296-000, Consolidation of Additional Siemens Breaker Issues
PER 03-010054-000, Problems Found During Performance of SI-266 Package P6451
PER 60199, Siemens Breaker Problems
PER 64674, RHR Pump 1A Did Not Start
WO 04-776671-005, Visual Inspection of "A" Train Breaker for Bradding Issues, RHR Pump 1A
WO 04-775027-000, Inspect Population of Spare Breakers for MOC Slide Problems Seen at Siemens
SI-266.1.1, Inspection of ITE 7.5HK-500 6900-V Breakers and Siemens 6900-V Vacuum Breakers, Revision 26, performed July to October 2003, P6451

Section 4OA5: Independent Spent Fuel Storage Installation

Procedures
O-SI-DCS-079-003.0 HI-Storm Average Surface Dose Rates, Revision 2
O-SF-DCS-079-001.0 HI-Storm System Site Transportation, Revision 0007
O-SI-DCS-079-002.0 HI-Trac Contamination Surveys, Revision 2

One Liner Survey Report Survey Nos. 071204-8, 071304-5, 071104-2, 071204-11, and 071204-9

LIST OF ACRONYMS

(Note: Ensure that acronyms listed are actually used in each report. For this sample report, all acronyms may not be listed here, or some may not be used in the report)

°F	degrees Fahrenheit
ALARA	as low as is reasonably achievable
ARMs	area radiation monitors
AOP	abnormal operating procedures
CCW	component cooling water
CFR	*Code of Federal Regulations*
COC	certificate of compliance
CR	condition report
DCN	design change notice
ECW	essential chill water
EGTS	emergency gas treatment system
EDG	emergency diesel generator
EOF	emergency operations facility
ERCW	essential raw cooling water
ESW	emergency service water
FSAR	Final Safety Analysis Report
GPS	global positioning system
GTM	gamma tool monitor
H_2O_2	hydrogen/oxygen
HPT	health physics technicians
JPM	job performance measure
LCP	loop control processor
LER	licensee event report
LOCA	loss of coolant accident
MOC	mechanism operated cell
MCD	maximum channel deviation
NCV	non-cited violation
NEI	Nuclear Energy Institute
NPF	nuclear power facility
NRC	U.S. Nuclear Regulatory Commission
OSP	Outage Safety Plan
PERs	problem evaluation reports
PCIV	primary containment isolation valve
PI	performance indicator
PORV	power-operated relief valve
RG	Regulatory Guide
RHR	residual heat removal
RHRSW	residual heat removal service water
RMOV	reactor motor-operated valve
RTP	rated thermal power
RPS	reactor protection system
RSS	remote shutdown system
RWP	radiation work permit
SCBA	self-contained breathing apparatus
SDP	significance determination process

SI	surveillance instructions
SLOCA	small break loss-of-coolant accident
SR	surveillance requirement
SRV	safety relief valve
SSC	structure, system, or component
TBD	to be determined
TLD	thermo luminescent dosimeter
TMI	Three Mile Island
TS	Technical Specification(s)
TSI	Technical Specification Interpretation
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item
WOs	work orders