EXHIBIT 3

SAMPLE REACTOR INSPECTION REPORT

The inspection report that follows 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 sufficient 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, summary of findings, and report details.

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.:	050998/2002007 and 0509992002007
Licensee:	Greckenshire Power & Light (GP&L)
Facility:	Dirojac Electric Station
Location:	10 Fourth Street Fridge, North Dakota
Dates:	December 30, 2001 - March 30, 2002
Inspectors:	***Note: Only inspectors who provided an input to the report*** J. Larkin, Senior Resident Inspector, Reactor Projects Branch B J. Henry, Resident Inspector, Reactor Projects Branch B J. Boyle, Senior Health Physicist, Plant Support Branch (Sections 2OS1, 2PS2) ***(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].)

CONTENTS

Note: Table of contents is only required if report detail exceeds 20 pages. Page numbers below may not exactly match the report depending on the printer used.

SUMMARY OF FINDINGS

IR 05000998/2002-007, 05000999/2002-007; 12/30/2001 - 03/30/2002; ***(Note: the dates of inspection come after the report #)*** Dirojac Electric Station, Units 1 and 2*;* Fire Protection, Maintenance Risk Assessment and Emergent Work, Operator Work-Arounds, Access Control to Radiologically Significant Areas, Event Followup, and Cross-Cutting Areas. ***(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. Six Green non-cited violations (NCVs), one Green finding, and one unresolved 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. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Initiating Events
(Note: Each finding is self-contained for PIM entry with respect to abbreviations)

- <u>Green</u>. The inspectors identified a non-cited violation of Technical Specification 5.4.1.a ***(regulation cited)*** because a surveillance procedure for calibrating a safety relief valve (SRV) while at power was not adequate. 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. ***(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. The finding is of very low safety significance because all mitigation systems were available during the use of the surveillance procedure. (Section 4OA3.1) ***(Note: Briefly describes why***

greater than minor, provides effect on cornerstone, and states why not greater than green.)

- Green. The inspector identified a finding because the work controls for an electrical splice associated with a main generator C-phase current transformer were inadequate and resulted in a reactor trip.
  This finding is greater than minor because it affected an attribute and the objective of the Initiating Events Cornerstone in that procedure inadequacies resulted in a perturbation in plant stability by causing a reactor trip. The finding is of very low safety significance because, although it caused a reactor trip, it did not increase the likelihood of a primary or secondary system loss of coolant accident initiator, did not contribute to a combination of a reactor trip and loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood (Section 4OA3.6).

Cornerstone: Mitigating Systems

- TBD. The inspector identified a violation having potential safety significance greater than very low significance because 8 of 12 fire protection sprinkler heads in the Unit 1 component cooling water pump room were not located in a manner required by GP&L's Fire Protection Review Report. The water spray pattern from eight sprinkler heads was partially obstructed by ventilation ductwork.

  This finding is unresolved pending completion of a significance determination. This finding is greater than minor because it is associated with Fire Protection equipment performance and degraded the ability to meet the cornerstone objective. The finding was determined to have potential safety significance greater than very low significance because of the large percentage of sprinklers that were obstructed, significantly reducing, their effectiveness. (Section 1R05.1) ***(Note: This URI is included as a finding in the cover letter because the significance may be greater than green)***

- Green. The inspectors identified a non-cited violation for the licensee's failure to comply with10 CFR 50, Appendix R, Criterion III.L.2.b. This violation is related to the unintended closure of both units' residual heat removal (RHR) pump minimum flow control valves, which could result in the failure of RHR pumps during some fire scenarios.

  This finding is greater than minor because it affected the Mitigating System Cornerstone objective of equipment reliability, in that closure of the minimum flow control valves could cause the RHR pumps to run dead-headed in the early stages of several fire scenarios, which could lead to pump failure. The finding is of very low safety significance because automatic and manual suppression systems were available, redundancy existed in the RHR system, two cables would have to be damaged in a particular manner to cause loss of function, and the plant power conversion system remained available through operator action to remove core heat. (Section 4OA5)

Cornerstone: Occupational Radiation Safety

- Green. The inspector identified an apparent violation of 10 CFR 20.1501(a) for failure of the licensee to adequately evaluate the radiological hazards associated with radioactive particles. The radioactive particles created a condition which, with a minor alteration, could have resulted in personnel being exposed substantially over the regulatory dose limit.

  This finding is greater than minor because if left uncorrected, it could result in a more significant safety concern. The finding is of very low safety significance because it did not constitute an as-low-as-reasonably-achievable (ALARA) finding, did not involve a very high radiation area (i.e., an area greater than 500 rem/hour), only affected the radiation safety cornerstone, and no overexposure is known to have occurred. This finding is a non-cited violation of 10 CFR 20.1501(a) (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.

- 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, 2002, 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, 2002, 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.

(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

<u>Summary of Plant Status</u> ***(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 planned reduction to 65 percent power on February 9, 2002, for maintenance on the 5A feedwater heater tube side drain. Unit 1 returned to 100 percent power on February 13, 2002. ***(Note: Power reduction included because of significant duration - if only a few hours it wouldn't be worth mentioning).***

Unit 2 was shut down for refueling at the beginning of the inspection period. On January 29, 2002, Unit 2 reached full RTP and operated at or near full RTP for the remainder of the inspection period with the exception of planned control rod pattern adjustments and control rod drive maintenance and testing. ***(Note: This disclaimer for normal plant down powers is optional)***

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity ***(Note: Only 1R CS's listed - EP is listed above the 1E section. If no EP section then do not include it here)***

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

a. <u>Inspection Scope</u> ***(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 January 31, 2002, the inspectors reviewed the licensee's preparations for inclement weather conditions. On January 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 reviewed documents listed in the Attachment ***(Note: Use Attachment when have more than a few reviewed documents)*** and 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 if operators could access controls and indications for those systems required for safe control of the plant.

b. <u>Findings</u> ***(Note: No findings of significance)***

No findings of significance were identified.

1R04 Equipment Alignments

a. Inspection Scope

Partial System Walkdowns. ***(Note: The 2 different types of alignment verifications in this IP are differentiated in the writeup)*** The inspectors performed two partial system walkdowns during this inspection period. ***(Note: Completed IP sample size clearly stated)*** On February 3, 2002, ***(Note: "When" because it's somewhat relevant-it's dependent on the day the other system was OOS)*** the inspectors walked down ***(How)*** one complete train of Unit 2 "B" turbine building closed cooling water (TBCCW) while the "A" TBCCW pump was out of service for maintenance ***(Note: A good "why" that is a required aspect of the IP).*** On March 4, the inspectors walked down Division II of the ESW system while the Division I ESW system was out of service for maintenance and testing. To evaluate the operability of the selected train or system when the redundant train or system was inoperable or out of service, the inspectors checked for correct valve and power alignments by comparing positions of valves, switches, and electrical power breakers to the procedures listed below ***(Acceptance criteria)*** as well as applicable chapters of the Final Safety Analysis Report (FSAR):

- OP-054-001, "Emergency Service Water System (ESW)"
- OP-215-001, "Turbine Building Closed Cooling Water System (TBCCW)"
- ON-215-001, "Loss of Turbine Building Closed Cooling Water (LTBCCW)"
- OP-244-001, "Condensate and Feedwater System (CD)"

(*Note: Do not need to use Attachment when have only a few reviewed documents)*

Complete System Walkdown. The inspectors conducted a detailed review of the alignment and condition of the Unit 1 essential chilled water (ECW) system. ***(Note: "When" not used since inspection known to be done this inspection period, and that's sufficient)*** 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 electrical power requirements, 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 within the setpoints.
- 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. ***(Note: Thoroughness of scope since no findings - it reflects main parts of inspection procedure that were done)***

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

On July 18, 1999, ***(Note: "When" used because relevant to finding)*** the inspector walked down ***(How)*** accessible portions of the six areas ***(Note: Required quarterly sample size for this IP)*** described below ***(What)*** with the fire protection engineer to assess the licensee's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. As part of the inspection, the inspectors reviewed ***(Acceptance criteria)*** the licensee's Fire Protection Review Report, Revision 9, dated August 8, 1997, ***(Note: Rev. & date included because used to support a finding)*** to ascertain the requirements for required fire protection design features, fire area boundaries, and combustible loading requirements for these areas. Documents reviewed during the inspection are listed in the Attachment.

- "C" emergency diesel generator (EDG) room while the room was being painted
- Unit 1 emergency switch gear rooms
- Unit 1 component cooling water (CCW) pump room
- Unit 2 CCW pump room

- Unit 1 standby liquid control system and reactor protection system (RPS) instrument rack areas, during modification of the fire detection system with pre-action system PA-151 out of service
- ESW system pump house during maintenance and testing

b. Findings ***(Note: Four-part format titles should be included as shown to clarify logic of format)***

Introduction. A finding was identified in that Unit 1 CCW pump room sprinklers were partially obstructed, having potential safety significance greater than very low significance. This is an unresolved item (URI) pending completion of the SDP.

Description. The inspectors noted that 10 of the 12 fire protection system sprinkler heads near and above the Unit 1 CCW pumps were partially obstructed by a ventilation duct. The site fire protection engineer stated that obstructions caused by ventilation ducts were acceptable under Section A-4-4.13 of the National Fire Protection Association (NFPA) 13-1978, "Standards for the Installation of Sprinkler Systems" because the ventilation ducts were less than 4 feet wide and, therefore did not require an additional sprinkler underneath the duct. However, the NFPA standard includes requirements for spacing sprinkler heads in relationship to obstructions, including ducts less than 4 feet wide. Section A-4-4.13 ultimately leads to Section 4-2.4.6 of the NFPA standard, which specifies the maximum allowable distance the sprinkler head can be from the side of an obstruction. Eight of 10 sprinklers did not meet the distance requirements of Section 4-2.4.6, in that they were all positioned above the bottom of the ventilation ducts and less than one foot distance from the side of the ducts. The licensee initiated CR 2000-1858 to evaluate the operability of the sprinkler system and possible corrective actions. Upon discovery of the distance problem, the licensee declared the Unit 1 CCW room fire suppression system inoperable, placed a permanent fire watch in the room and initiated the development of a modification to modify the sprinkler locations.

Analysis. ***(Note: What set of conditions make the finding greater than minor)*** The finding adversely impacted fire suppression equipment reliability and capability. ***(Note: Which cornerstone was affected)*** Because the finding affected the reactor safety mitigating system cornerstone objective, the finding is greater than minor. The finding also was determined to have potential safety significance greater than very low significance because of the large percentage of sprinklers that were obstructed (greater than 25%) and that the automatic suppression capability could be reduced significantly. ***(Note: assumption used.)***

Enforcement. The licensee's Fire Protection Review Report commits to NFPA 13-1978. Section 4-4.2.6 states that the maximum sprinkler head distance above an obstruction is zero inches, when the sprinkler is located less than one foot from the obstruction. Contrary to the above, 8 of 12 sprinkler heads in the Unit 1 CCW pump room did not meet this requirement. Pending determination of the finding's safety significance, this finding is identified as URI 50-998/02-07-01, Suppression Sprinklers in CCW Pump Room Partially Obstructed. ***(Note: Because the significance of this finding has not been determined, it should be "counted" in the cover letter as a potentially "greater-than-Green" item, and described in the summary of findings with a TBD color)***

1R14 Personnel 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, 2002, 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, 2002, 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, 2002.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspector's reviewed both of the operability determinations the licensee had generated that warranted selection on the basis of risk insights. ***(Note: Justifies why less than normal sample size of 3/qtr)*** The selected samples are addressed in the CRs ***(What)*** listed below. The inspectors assessed the accuracy of the evaluations, the use and control of compensatory measures if needed, and compliance with the TS. The inspectors review included a verification that the operability determinations were made as specified by Procedure NDAP-QA-0703, "Operability Assessments." The technical adequacy of the determinations was reviewed and compared ***(Acceptance criteria)*** to the TS, Technical Requirements Manual, FSAR, associated design-basis documents, Procedure ST 83308-C, "Testing of Safety-Related NSCW System Coolers and Request for Engineering Assistance REA 01-V2A134, Effect on 2B SIP with Lube Oil Cooler in a 2-pass Arrangement."

- CR 29934 Main steam SRV eductor changed during rebuild without an evaluation
- CR 34968 Unit 2 "C" RHR pump discharge check valve leaks

b. Findings

No findings of significance were identified.

1R20 Refueling and 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, 2002, 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. ***(Note: Scope should be a complete but <u>concise</u> listing of the required IP activities)***

- 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.

Cornerstone. Emergency Preparedness ***(Note: EP Cornerstone listed above the 1E section - if no writeup in EP then do not list in report)***

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety
(NOTE: Public Radiation Safety Cornerstone listed above 2P Sections. If no 2P Sections, then do not list in report)

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope ***(Note: Example of a finding covered by a non-risk-informed SDP)***

The inspectors reviewed the controls for assuring radiological safety associated with waste consolidation and shipping of irradiated reactor hardware. ***(NOTE: Wherever report is "obvious" that one inspector did review, use "inspector". Use inspectors everywhere else)*** The inspectors observed the control of radioactive particles, attended an as-low-as-reasonably-achievable (ALARA) pre-job meeting for removal of a satellite roller punch tool, and discussed the controls with members of the site radiation protection staff. The inspectors also reviewed the following types of documents: refueling floor radiation work permits (RWPs), radiation surveys, and personnel dose evaluations from exposure to radioactive particles. Procedures for radioactive particle controls and resulting skin dose assessments were also reviewed. In addition, 10 CRs associated with radioactive particle events since December 2001 were reviewed, including CR 289925, dated February 20, 2002, that included a Level 1 root cause analysis of radiological controls. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

Introduction. A Green NCV was identified for failure to adequately evaluate the radiological hazards associated with radioactive particles, as prescribed in 10 CFR 20.1501(a).

Description. The NRC-identified that, during the period between September 2, 2001, and February 28, 2002, the licensee failed to adequately evaluate the radiological hazards associated with disposing of irradiated reactor hardware, equipment and tools that were contaminated with highly radioactive particles (i.e., very small particles, principally cobalt-60, containing millicurie levels of radioactivity).

During this period, the licensee had set up a containment tent on the refueling floor to provide an enclosure for working on irradiated tools and equipment. Radiological controls for the work required protective clothing and established that individuals would be surveyed at 30 minute intervals to determine if they had picked up any radioactive particles. On January 7, 2002, a decontamination worker was scrubbing material off a machine stand inside the enclosure. When the individual was surveyed at a regular interval, the licensee located a highly radioactive particle on the individual's boot. Subsequently the licensee determined that the radioactive particle contained about 1.75 millicuries of Cobalt-60. A conservative dose assessment by the licensee indicated the worker received a 17 rem shallow-dose equivalent (SDE) exposure, which was within the 10 CFR 20.1201 shallow and extremity dose limit of 50 rem.

On January 12, 2002, after the licensee found a 5 millicurie particle on the refueling floor, the licensee wrote a Level 1 CR and assigned an Event Review Team (ERT) to evaluate the condition. Following the ERT's recommendation, the licensee reduced the stay-time interval to 15 minutes. Subsequently, other highly radioactive hot particles were discovered on June 25 (two hot particles, 2.6 millicuries each) and February 2, 2002, (7.8 millicuries) in areas accessible to workers. While these discoveries provided opportunities to re-evaluate the radiological hazard potential, the licensee continued to implement only the established radiological controls and failed to recognize or consider any other radiological implication or exposure hazard posed by the recurring highly radioactive particles.

The 1.75 millicurie hot particle, which caused the 17 rem SDE dose to the worker on January 7, was capable of causing an SDE dose to the worker in excess of the occupational limits of 50 rem, well within the licensee's 15 minute survey interval, if the particle had been located on a less protected portion of the body. The actual dose to the worker was limited because the particle was on the outside of the worker's boot. The other radioactive particles encountered by the licensee had the potential to exceed both the SDE and deep dose equivalent (DDE) limits within several minutes if located on a portion of the body other than an extremity, a potentiality that could occur with only a minor change in the circumstances. Accordingly, the radioactivity exhibited by these particles was sufficient to potentially cause personnel exposures in excess of the regulatory requirements.

<u>Analysis</u>. ***(Note: Explanation of significance)*** The inspector determined that GP&L's failure to adequately evaluate the radiological hazards associated with radioactive particles is a performance deficiency because GP&L is expected to meet the requirements of 10 CFR 20.1501(a). Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements or GP&L procedures. This finding is greater than minor because it is associated with program and process attributes and affected the objective of the Radiation Safety Cornerstone to protect worker from exposure to radiation. In addition, if left uncorrected, this finding could result in a more significant safety concern (i.e. an actual overexposure). ***(Note: Assumption)*** This finding was evaluated using the Occupational Radiation Safety SDP and was preliminarily determined to be a finding of low to moderate safety significance. The Occupational Radiation Safety SDP defines a substantial potential for overexposure attributable to hot particles. In this case the radioactivity of the particles encountered was sufficiently high that the substantial potential for over exposures from the particles existed not only for the skin (i.e., SDE), but also for the whole body, i.e.,DDE. Such DDE exposures could have exceeded the 5 rem annual total effective dose equivalent limit in 10 CFR 20.1201.

<u>Enforcement</u>. 10 CFR 20.1501(a) states, in part ***(Note: Inspectors must state the requirement)*** "Each licensee shall make or cause to be made surveys that (1) may be necessary for the licensee to comply with the regulations in this part and (2) are reasonable under the circumstances to evaluate... (ii) concentrations or quantities of radioactive materials..." ***(Note: Inspectors must state how the requirement was violated)*** Contrary to the above, the licensee's failure to effectively evaluate the radiological hazard presented by these radioactive particles relative to DDE exposure resulted in a condition which, with a minor alteration, could have resulted in an exposure of personnel substantially in excess of the regulatory dose limit. Because the failure to effectively evaluate the radiological hazard is of very low safety significance and has been entered into the CAP (CR 902432), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-998, 999/02-07-05, Failure to Adequately Evaluate Radioactive Particle Radiological Hazards.

(Note: If there are PS section(s), insert the following here)
Cornerstone: Public Radiation Safety

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the period from April 2001 through March 2002. 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," Rev. 1, were used to verify the basis in reporting for each data element.

Reactor Safety Cornerstone
(Note: Aggregate PIs by cornerstone to simplify/shorten repetitive writeups)

- Unplanned Scrams per 7,000 Critical Hours
- Scrams with a Loss of Normal Heat Removal
- Unplanned Transients per 7000 Critical Hours

The inspector reviewed a selection of LERs, portions of Unit 1 and Unit 2 operator log entries, daily morning reports (including the daily CR descriptions), the monthly operating reports, and PI data sheets to determine whether the licensee adequately identified the number of scrams and unplanned power changes greater than 20 percent that occurred during the previous four quarters. This number was compared to the number reported for the PI during the current quarter. The inspectors also reviewed to verify the accuracy of the number of critical hours reported and the licensee's basis for crediting normal heat removal capability for each of the reported reactor scrams. In addition, the inspectors also interviewed licensee personnel associated with the PI data collection, evaluation, and distribution.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness PI

Licensee records reviewed included those used by the licensee to identify occurrences of locked high-radiation areas, very high-radiation areas, and unplanned personnel exposures. Additional records reviewed included ALARA records addressing individual exposures. The inspectors also interviewed licensee personnel that were accountable for collecting and evaluating the PI data.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

(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 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. Annual Sample Review

a. Inspection Scope

The inspectors selected two deficiency reports ***(Note: Make clear that 2 samples are being used)*** for detailed review (DRs 2001-35 and 2001-568). The deficiency reports were associated with a failed governor for the turbine-driven auxiliary feedwater pump and broken manifold bolts on the EDGs. The reports were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors evaluated the reports against the requirements of the licensee's CAP as delineated in Site Administration Procedure SAP-00-001, Problem Identification and Correction, and 10 CFR 50, Appendix B ***(Acceptance criteria)***.

b. Findings and Observations ***(Note: Change the "Findings" section title here to reflect that the following information is probably not a finding)***

(Note: The expectation is to document the insights from the annual sample review even if no findings. Only factual information relative to the inspection requirements may be documented for these samples to allow for integration into the biennial PI &R assessment. Assessments of overall PI&R effectiveness should only be made during the biennial inspection.)
There were no findings identified ***(Note: Make it clear you are not creating a finding even though documenting in the Findings section)*** associated with the two reviewed samples; however, the inspectors identified that the licensee failed to complete a root cause evaluation within the time frame called for by Procedure SAP-00-001. This procedure requires that a root cause evaluation be

performed within 90 days of classifying the issue as a Category 1 event by the Corrective Action Review Board. However, the licensee's root cause evaluation was not completed until 197 and 202 days respectively for each DR after classification as a Category 1 event. The inspectors also identified several other minor examples where format and interfacing plant department review requirements of SAP-00-001 were not completed. ***(Note: This sets up several examples/data points that show that the licensee does not routinely comply with all the requirements of their CAP. This can be used during the biennial if more examples identified to make a broader PI&R program assessment conclusion)*** The inspectors verified that the root cause evaluation and associated corrective actions were appropriate and also timely, relative to the identified problem; therefore no violation of regulatory requirements or findings were identified.

2. <u>Cross-References to PI&R Findings Documented Elsewhere</u>

Section 1R16 describes that the licensee had identified a problem with RHRSW radiation monitor alarms during a system self-assessment, but had not tracked the concern effectively in their procedure change program and had not entered the specific problems from the self-assessment in the CAP. Consequently the alarm problems were never fully evaluated or corrected.

Section 4OA3 of the report describes a finding for an inadequate surveillance procedure for calibrating a safety relief valve, that could have been reasonably identified during previous testing, if licensed operators and instrumentation technicians had prepared a deficiency report

Section 1R22 describes a finding for failure to demonstrate that "required pressurizer heaters are capable of being powered from an emergency power supply" every 18 months. The licensee had a previous opportunity to identify the finding.

Section 2OS1 describes a finding for failure to adequately evaluate the radiological hazards associated with radioactive particles. Since the licensee failed to recognize or consider other radiological implications or exposure hazards posed by the recurring highly radioactive particles, the finding is indicative of a potential deficiency in the licensee's corrective action program within the radiation protection program.

4OA3 <u>Event Followup</u>

1. Inadvertent SRV Opening During Testing

a. <u>Inspection Scope</u>

The inspectors observed control room personnel responding to an unexpected opening of an SRV on February 18, 2002. 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. <u>Findings</u> ***(Note: Inspection results first)***

<u>Introduction</u>. 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.

<u>Description</u>. On February 18, 2002, a self-revealing finding was identified ***(Note: Finding is self-revealing, not licensee-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. This problem identification concern is referenced in Section 4OA2.

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 non-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 mall LOCA increased, the exposure time for this condition was less than 30 days and all mitigation capabilities described on the SDP phase 2 worksheet for SLOCA core damage sequences were maintained. ***(Note: This is a simple example of a finding that is green based on the results of an SDP phase 2 evaluation.)***

Enforcement. TS 5.4.1.a requires written procedures be established, implemented, and maintained covering the activities specified in ***(Note: Requirement stated)*** 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, 2002, resulted in an inadvertent opening of an SRV during testing. ***(Note: How requirement was violated)*** 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, 999/2002007-07, Failure to Maintain Adequate Surveillance Instruction to Prevent Inadvertent SRV Opening.

(Note: Example of closing an LER, below)

2. (Closed) LER 05000999/2002003-01, Inadvertent Engineered Safety Feature Actuation Caused by Loss of RPS Power Supply

On February 4, 2002, 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. 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)

3. (Closed) LER 05000998, 999/2002009-02 Primary Containment Isolation Valves not Checked per Surveillance Requirements

On March 10, 2002, 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-ID NCV, below)

4. (Closed) LER 05000999/2002004-04. Technical Specification Interpretation Incorrect — Operation Prohibited by TS

On February 17, 2002, 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, 2002, 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 LIV, only significance of LER issue discussed here. Section simply refers to 4OA7 for enforcement)

5. (Closed) LER 05000998/2002014-02, Automatic Turbine/Reactor Trip Due To Main Transformer Protection Circuit Ground Due To Inadequate Cable Splice

(Note: Use 4-point format if have a finding)

a. Inspection Scope

The inspectors reviewed the LER and CR 2002-1858, which documented this event in the corrective action program, to verify that the cause of the July 13, 2002, Unit 1 reactor trip event was identified and that corrective actions were reasonable. The turbine trip/reactor trip was caused by a bolted cable splice associated with a C-phase main transformer differential relay shorting to a junction box. The inspectors reviewed plant parameters and verified that timely notifications were made in accordance with 10 CFR 50.72, that licensee staff properly implemented the appropriate plant procedures, and that plant equipment performed as required.

b. Findings

Introduction. A Green self-revealing finding for inadequate work instructions was identified.

Description. The licensee determined, pending confirmatory testing, that the root cause of the event was inadequate work instructions that allowed lower temperature-rated tape to be used on a cable replacement splice because the operating environmental conditions within the junction box were unknown and/or inadequate application of splice material.

Analysis. The inspectors determined that the finding is a performance deficiency because GP&L administrative procedure (?????) requires adequate work instructions for a cable replacement splice. This finding was greater than minor because it affects an attribute and objective of the Initiating Events Cornerstone in that procedure inadequacy resulted in a perturbation in plant stability by causing a reactor trip. The finding was assessed using the Significance Determination Process for Reactor Inspection Findings for At-Power Situations and was determined to be of very low safety significance (Green). While the finding resulted in an actual trip, the inspectors determined that the finding did not contribute to the likelihood of a primary or secondary system LOCA initiator, did not contribute to a loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood. This finding is in GP&L's corrective action program as CR 2002-1858.

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because it occurred on non-safety-related secondary plant equipment.

4OA5 Other (Optional 4-point format)

(Closed) URI 05000998, 999/2001005-01: Residual Heat Removal Pump Minimum Flow Control Valve Appendix R Requirements ***(Note: Non-LER open items normally closed in Section 4OA5)***

Introduction. A Green NCV was identified for failure to comply with 10 CFR 50, Appendix R, Criterion III.L.2.b, related to the unintended closure of both units' RHR pump minimum flow control valves

Description. During the triennial fire protection inspection (NRC Inspection Report 05000998, 999/2001005, dated August 8, 2001), the inspectors identified a finding having potential safety significance greater than very low significance, involving the potential unintended closure of RHR pump minimum flow control valves due to fire damage to control cables. The inspectors had determined that the plant condition did not meet the requirements of 10 CFR 50, Appendix R, Criterion III.L.2.b and that it was a condition not covered by the plant's operating and emergency procedures. Criterion III.L.2.b states that, when using "alternative" shutdown in response to a fire, one performance goal for the shutdown functions is that the reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core. Closure of the minimum flow control valves could cause the RHR pumps to run dead-headed in the early stages of several fire scenarios, which could lead to pump failure. Hence the issue impacted the ability to safely shutdown the plant for the credible initiating event of fire due to loss of function of a mitigating system, specifically the RHR system. The significance of this finding had not been determined at the conclusion of the inspection.

Analysis. The inspectors had determined that the finding was associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of equipment reliability. Therefore, the finding is greater then minor. The inspectors noted that reliance on RHR appeared to be an important part of the licensee's safe shutdown analysis.

During the current inspection period, the inspectors determined that the issue was of very low safety significance (Green). Some of the factors (assumption used in the SDP, IMC 0609, Appendix F) causing the issue to be of very low safety significance were:

- Automatic and manual suppression systems were available which reduced the estimated likelihood rating
- Due to redundancy in the RHR system two cables would have to be damaged in a particular manner to cause loss of function
- The plant power conversion system, which was not included in the safe shutdown analysis, remained available through operator action to remove core heat

Enforcement. Because this failure to comply with 10 CFR 50, Appendix R, Criterion III.L.2.b is of very low safety significance and has been entered into the CAP (CR 06-4567), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000998, 999/2002007-08, Failure to Meet 10 CFR 50, Appendix R Requirements for RHR Minimum Flow Control.

4OA6 Meetings, including Exit

On March 29, 2002, the resident inspectors presented the inspection results to Mr. B. Handcuff and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

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, 2002, 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, 2002, 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.

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, 999/2002007-01	URI	Suppression Sprinklers in CCW Pump Room Partially Obstructed (Section 1R05)
05000998,999/2002007-02	AV	Failure to Adequately Evaluate Radioactive Particle Radiological Hazards (Section 20S1)

Opened and Closed

05000998, 999/2002007-04	NCV	Failure to Demonstrate Pressurizer Heaters Emergency Power Capability (Section 1R22)

05000998, 999/2002007-02	NCV	Failure to Adequately Evaluate Radioactive Particle Radiological Hazards (Section 20S1)
05000998, 999/2002007-03	NCV	Failure to Maintain Adequate Surveillance Procedure to Prevent Inadvertent SRV Opening (Section 4OA3.1)
05000998/2002007-04	FIN	Inadequate Work Control Resulted in a Reactor Scram (Section 4OA3.6)
05000998, 999/2002007-04	NCV	Failure to Meet 10 CFR 50, Appendix R Requirements for RHR Minimum Flow Control Valves (Section 4OA5)

Closed

05000998, 999/2001005-01	URI	Residual Heat Removal Pump Minimum Flow Control Valve Appendix R Requirements (Section 4OA5)
05000999/2002003-01	LER	Inadvertent Engineered Safety Feature Actuation Caused by Loss of RPS Power Supply (Section 4OA3.2)
05000998, 999/2002009-02	LER	Primary Containment Isolation Valves not Checked per Surveillance Requirements (Section 4OA3.3)
05000999/2002004-04	LER	Technical Specification Interpretation Incorrect — Operation Prohibited by TS (Section 4OA3.5)
05000998/2002014-02	LER	Automatic Turbine/Reactor Trip Due To Main Transformer Protection Circuit Ground Due To Inadequate Cable Splice (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)

Section 1R01: Adverse Weather Protection

Procedures

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

Section 1R05: Fire Protection

Procedures

FP 92845-2, Zone 145 - NSCW Cooling tower 2A, Mechanical and Electrical Tunnels 2T2A, 2T3A and 2T5A Fire Fighting Preplan
FP 92860A-2, Zone 160A - NSCW Pumphouse Train A Fire Fighting Preplan
FP 92732-1, Zone 32 - Auxiliary Building - Level B Firefighting Preplan
FP 92861-2, Diesel Generator Building Fire Fighting Preplan
FP 92863-2, Diesel Generator Building Train A DFO Day Tank, Zone 163, Fire Fighting Preplan
FP 92865-2, Diesel Generator Tanks and Pumphouse Zone 165- Fire Fighting Preplan
FP 92866-2, Diesel Generator Tanks and Pumphouse Zone 166- Fire Fighting Preplan
FP 92705-2, Zone 5 - Auxiliary Building - Level D, Containment Spray Pump B Firefighting Preplan
FP 92720-1, Zone 20- Auxiliary Building CVCS Pump RM Train A Firefighting Preplan

Section 1R19: Post-Maintenance Testing

Procedures

PMT 14430-2, TBCCW Cooling Tower Fans Monthly Test
PMT 14808-1, C EDG Check Valve IST and Response Time Test
PMT 83308-C, Testing of Safety-Related TBCCW System Coolers

PMT 14980-2, Diesel Generator Operability Test
PMT 25093-C, Reactor Building Chill Water Hardware Replacement
PMT 28705-C, Reactor Building Chill Water Circuit Breaker Inspection and Testing

Section 1R20: Refueling and 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 2OS1: Access Control to Radiologically Significant Areas

Procedures

SPP-1.1, Training and Qualification of Personnel
SPP-5.1, Radiological Controls
Training -20, Health Physics Technician-Training
SPP-5.1 Radiological Controls
Radiological Control Department Procedure (RCP)-1 Conduct of Radiological Controls
RCP-3 Administration of Radiation Work Permits
Radiological Control Instruction (RCS) - 100 Control of Radiological Work
RCS-111 Special Exposure Monitoring
RCS-112 Operation and Calibration
Technical Instruction - 7.005 Storage of Material in the Spent Fuel Pool, Cask Pit, & New Fuel Vault

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

ADAMS	Agency-Wide Documents Access and Management System
ALARA	as low as is reasonably achievable
CCW	component cooling water
CD	condensate and feedwater
CFR	*Code of Federal Regulations*
CR	condition report
DDE	deep-dose equivalent
EDG	emergency diesel generator
EOF	emergency operations facility
ERT	Event Review Team
ESW	emergency service water
FSAR	Final Safety Analysis Report
H_2O_2	hydrogen/oxygen
HPCI	high pressure coolant injection
LER	licensee event report
LOCA	loss of coolant accident
LTBCCW	loss of turbine build closed cooling water
NCV	non-cited violation
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
OSP	Outage Safety Plan
PARS	Publicly Available Records Systems
PCIV	primary containment isolation valve
PI	performance indicator
QA	quality assurance
RHR	residual heat removal
RPS	reactor protection system
RWP	radiation work permit
SDE	shallow dose equivalent
SDP	significance determination process
SR	surveillance requirement
SRV	safety relief valve
SSC	structure, system, or component
SI	surveillance instructions
TBCCW	turbine building closed cooling water
TS	Technical Specification(s)
TSI	Technical Specification Interpretation
URI	unresolved item