May 3, 2005

Mr. James A. Spina
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000220/2005002 and 05000410/2005002

Dear Mr. Spina:

On March 31, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Nine Mile Point Nuclear Station (NMPNS), Units 1 and 2. The enclosed integrated inspection report (IR) documents the inspection findings which were discussed on April 15, 2005, with Mr. Tim O'Connor and other members of your staff.

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

This report documents four NRC-identified findings of very low safety significance (Green). Three of the findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because the violations were entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this IR, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001 with copies to the Regional Administrator Region I, the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Nine Mile Point.

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

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Sincerely,

/RA/

James M. Trapp, Chief
Projects Branch 1
Division of Reactor Projects

Docket Nos.: 50-220, 50-410
License Nos.: DPR-63, NPF-69

Enclosure: Inspection Report 05000220/2005002 and 05000410/2005002
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	50-220, 50-410
License Nos.:	DPR-63, NPF-69
Report No.:	05000220/2005002 and 05000410/2005002
Licensee:	Nine Mile Point Nuclear Station, LLC (NMPNS)
Facility:	Nine Mile Point, Units 1 and 2
Location:	348 Lake Road Oswego, NY 13126
Dates:	January 1, 2005 - March 31, 2005
Inspectors:	G. Hunegs, Senior Resident Inspector B. Fuller, Resident Inspector E. Knutson, Resident Inspector J. Furia, Senior Health Physicist D. Silk, Senior Emergency Preparedness Inspector
Approved by:	James M. Trapp, Chief Projects Branch 1 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000220/2005002, 05000410/2005002; 01/01/05 - 03/31/05; Nine Mile Point, Units 1 and 2; Equipment Alignment, Maintenance Risk Assessment, Surveillance Testing, and Event Follow-up.

This report covered a 3-month period of inspection by resident inspectors, and an announced inspection and an in-office review, by two region based inspectors. Three Green non-cited violations (NCVs), and one Green finding, 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. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding regarding an improperly installed flexible coupling in the Unit 2 high pressure core spray (HPCS) system suction line from the condensate storage tank (CST). The tie rods were not properly adjusted, thereby increasing its probability of failure during a seismic event. The performance deficiency is that an inadequate maintenance procedure had been prepared and used to install the HPCS CST suction line flexible coupling. As a result, the tie rods had not been adjusted in accordance with the vendor's specifications.

  The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance in accordance with phase 1 of the SDP because it was not a design or qualification deficiency, did not represent a loss of the HPCS system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. (Section 1R04)

- Green. The inspectors identified a NCV of 10 CFR 50.65(a)(4) for the failure to adequately manage the increase in risk that resulted from maintenance on the Unit 2, Division 2, 125 VDC battery (2BYS*BAT2B). Specifically, the sizing of fasteners was not adequately determined prior to installing a jumper around one of the battery cells, which resulted in the plant being maintained in a high risk configuration for approximately twice as long as would otherwise have been necessary. The performance deficiency associated with this event is failure to adequately plan the jumper installation for battery 2BYS*BAT2B cell 21, such that the time spent in a high risk plant configuration would be minimized.

  The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance in accordance with phase 1 of the SDP because it was not a design or qualification deficiency, did not represent actual loss of safety function of a single train for greater than its Technical Specification (TS) allowed outage time, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

The failure to adequately manage the increase in risk that resulted from the battery maintenance is an example of a cross-cutting issue in human performance at the organizational level. Specifically, the Engineering Department did not apply rigor commensurate with the sensitivity of the maintenance activity when they failed to determine the precise length of the required fasteners in developing the temporary change package (TCP); and, Maintenance personnel inappropriately excluded parts that were specified in the TCP when preparing for the activity, based on unavailability rather than technical justification. (Section 1R13)

- Green. The inspectors identified a non-cited violation (NCV) of Unit 1 TS 6.4.1.a concerning an inadequate procedure review and approval process related to the development of procedure N1-ST-V19, "Emergency Cooling System - Heat Removal Capability Test at High Power." Specifically, the licensee incorrectly determined that all aspects of the activity were controlled by other processes, thereby negating the requirement for a 10 CFR 50.59 screen. Subsequently it was determined that the procedure also contained changes that affect operation and control of other systems and therefore that a 10 CFR 50.59 screen should have been completed. The performance deficiency associated with this event is a failure to perform a 10 CFR 50.59 screen when one was required.

  The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affected the associated cornerstone objective of ensuring the capability of the emergency condenser system, a core decay heat removal system, to respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance in accordance with phase 1 of the SDP because it was not a design or qualification deficiency, did not represent an actual loss of the emergency condenser system safety function, and was not potentially risk significant due to seismic, flood, fire or weather related initiating events. (Section 1R22)

- Green. The inspectors identified a non-cited violation (NCV) of 10 CFR 50.65(a)(4) for the failure to adequately assess the increase in risk that resulted from maintenance on the Unit 1 control room ventilation system. Specifically, no assessment of risk was performed prior to opening doors which served as barriers between the mild environment of the control room and the potential harsh environment of the Turbine Building resulting from a high energy line break (HELB). The performance deficiency associated with this event is failure to adequately assess the increased risk from a HELB in the Turbine Building with doors in the HELB boundary open to the Control Room.

  The finding is more than minor because if left uncorrected, it would become a more significant safety concern in that actions to assess and manage increases in risk may not have been implemented. The finding was determined to be of very low safety significance in accordance with phase 3 of the SDP because it resulted in a change in core damage frequency (CDF) significantly below the green/white risk threshold. (Section 4OA3)

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Nine Mile Point Unit 1 (Unit 1) began the inspection period operating on five recirculation loops at 100 percent power. On March 7, Unit 1 automatically scrammed as the result of a turbine trip. The turbine trip was caused by a false high level signal in a reheater drain tank. The level indicator was repaired and the unit restarted on March 8, with full power attained on March 10. A normal reactor shutdown was performed on March 21, to commence refueling outage 18. The inspection period ended with Unit 1 shutdown for refueling.

Nine Mile Point Unit 2 (Unit 2) began the inspection period at 100 percent power. On January 15, power was reduced to 55 percent to support switching the operating and standby main feedwater pumps. On March 19, power was reduced to 55 percent for a planned control rod pattern adjustment. The power reduction was also utilized to support switching the operating and standby main feedwater pumps, perform steam system valve testing, and perform single control rod scram time testing. Power was returned to 100 percent the following day. On March 25, power was again reduced to 55 percent to support switching the operating and standby main feedwater pumps. The unit operated at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope (71111.01 - 1 Sample - cold weather preparations)

The inspectors examined three risk significant systems in the Unit 1 Reactor and Turbine Buildings to verify that design features, operating procedures, and in-plant conditions supported operation of these systems during periods of cold weather. Unit 1 documents reviewed included the Unit 1 Final Safety Analysis Report (FSAR), the Unit 1 Individual Plant Examination (IPE) for External Events, N1-OP-64, “Meteorological Monitoring,” N1-PM-A5, “Cold Weather Preparation and Operation,” and EPIP-EPP-26, “Natural Hazard Preparation and Recovery.” The following systems were examined:

- Core Spray System
- Containment Spray System
- Turbine Building Ventilation System

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdown. (71111.04 - 4 Samples)

The inspectors performed partial system walkdowns to verify proper system and component alignment and to note any discrepancies that would impact system operability. The walkdowns included control room switch and indication verification, physical inspection, and partial verification of the system lineup.

- On January 27, the inspector performed a partial system walkdown on the Unit 2 service water (SW) system based on safety significance. Procedure N2-OP-11, "Service Water System," was used for this review.

- On March 1, the inspector performed a partial system walkdown on the Unit 2 residual heat removal (RHR) subsystem C based on safety significance. Procedure N2-OP-31, "Residual Heat Removal System," was used for this review.

- On March 23, the inspectors performed a partial system walkdown on the Unit 1 shutdown cooling system due to increased risk significance during the refueling outage. Procedure N1-OP-4, "Shutdown Cooling System," was used for this review.

- On March 30, the inspectors performed a partial system walkdown on the Unit 1 spent fuel cooling system due to increased safety significance following full core offload and installation of a temporary heat removal system. Procedure N1-OP-6, "Fuel Pool Filtering and Cooling System," and Temporary Change Package N1-04-165, "SFC Temporary Cooling System for RFO18," were used for this review.

Complete System Walkdown. (71111.04S - Followup)

The inspectors reviewed the licensee's response to observations that had been made during a full system walkdown of the Unit 2 high pressure core spray system, as discussed in the previous NRC Integrated Inspection Report, 05000410/2004005.

b. Findings

Introduction. The inspectors identified a Green finding at Unit 2, when they found that a flexible coupling in the high pressure core spray system suction line from the condensate storage tank had not been properly installed, such that its probability of failure during a seismic event was increased.

Description. During a walkdown of the high pressure core spray (HPCS) system, the inspectors noted that a flexible coupling in the suction line from the CST did not appear to be installed correctly. Specifically, the four tie rods were set to allow what appeared to be too much axial travel, and also were not set at equal lengths. The licensee investigated the condition and determined that the tie rods had not been properly adjusted following replacement of the coupling during the 2004 refueling outage. They further concluded that the piping supports on either side of the coupling were adequately robust to ensure that the condition did not threaten the functionality of the coupling. The licensee noted that the coupling was not within the safety-class boundary of the HPCS system, and therefore did not perform a formal operability determination. The issue of improperly adjusted tie rods on the HPCS CST suction line flexible coupling was entered in the licensee's corrective action program as Deviation/Event Report (DER) NM-2005-54.

Analysis. The performance deficiency associated with this event was that an inadequate maintenance procedure had been prepared and used to install the HPCS CST suction line flexible coupling. As a result, the tie rods had not been adjusted in accordance with the vendor's specifications. The finding was greater than minor because it is associated with the Mitigating System Cornerstone attribute of mitigating equipment performance and affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Although the suppression pool is the safety-related source of water for the HPCS system, the CST serves as a reserve supply. Furthermore, since the HPCS system is normally aligned to take a suction from the CST, loss of this flow path would challenge the system by causing an automatic suction transfer to occur. Rupture of the HPCS CST suction line flexible coupling would represent a reduction in system reliability. The finding was determined to be of very low safety significance (Green) in accordance with Phase 1 of the Reactor Safety Significance Determination Process (SDP) because it was not a design or qualification deficiency that had been confirmed to result in a loss of function per Generic Letter 91-18, did not represent a loss of safety function, did not represent actual loss of safety function of a single train for greater than its Technical Specification (TS) allowed outage time, did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk-significant per 10 Code of Federal Regulations (CFR) 50.65 for greater than 24 hours, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

Enforcement. The HPCS CST suction line flexible coupling is outside of the safety class boundary of the HPCS system, and therefore is not subject to the requirements of 10 CFR 50, Appendix B. Consequently, no violation of regulatory requirements occurred. FIN 50-410/2005002-01, Improper Installation of HPCS Suction Line Flexible Coupling Due To Inadequate Procedure.

1R05 Fire Protection

a. Inspection Scope (71111.05Q - 12 Samples)

The inspectors walked down accessible portions of fire areas described below to assess the licensee's control of transient combustible material and ignition sources, fire detection and suppression capabilities, and fire barriers and any related compensatory measures. The condition of fire detection devices, and readiness of sprinkler fire suppression systems and fire doors, were also inspected against industry standards. In addition, the fire protection features were inspected, including ventilation system fire dampers, structural steel fire proofing, and electrical penetration seals. Reference material reviewed for installed features included the Unit 1 FSAR and the Unit 2 Updated Safety Analysis Report (USAR).

- Unit 1 Reactor Building (RB) Emergency Core Cooling System (ECCS) northeast Corner Room 198 ft
- Unit 1 RB ECCS southeast Corner Room 198 ft
- Unit 1 RB ECCS northwest Corner Room 198 ft
- Unit 1 RB ECCS southwest Corner Room 198 ft
- Unit 1 RB 261 ft
- Unit 1 Turbine Building northwest 305 ft
- Unit 2 RB 175 ft
- Unit 2 Standby Gas Treatment Rooms
- Unit 2 RB 215 ft
- Unit 2 RB 261 ft
- Unit 2 Division 1-3 Cable Chase Rooms, Control Building 288 ft
- Unit 2 Division 3 Emergency Diesel Generator (EDG) Room

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope (71111.06 - 1 Sample - Internal)

The inspectors performed a walkdown of the Unit 2 Diesel Generator Building to examine its susceptibility to internal flooding. This area was considered to be potentially risk significant due to it containing a large amount of safety class electrical switchgear, the station emergency electrical power sources, along with a potential major flood source, the SW system. Documents reviewed included the USAR, the Individual Plant Examination (IPE), and procedure N2-OP-66, "Miscellaneous Drains."

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope (71111.11Q - 2 Samples and 71114.06 - 2 Samples)

The inspectors reviewed two licensed operator requalification training activities (one per unit), to assess the licensee's training program effectiveness. The inspectors observed Unit 1 and 2 licensed operator simulator training on February 28. The inspectors reviewed performance in the areas of procedure use, self-checking and peer-checking, completion of critical tasks, and training performance objectives. Following the simulator training, the inspectors reviewed simulator fidelity through a sampling process. During the training, the inspectors evaluated emergency response organization (ERO) performance regarding initial and subsequent actions by licensed operators.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope (71111.12Q - 1 Sample)

The inspectors reviewed the performance and condition history of one high safety significant system, the Unit 2 emergency (safety-related) uninterruptible power supply (UPS) system (system VBA), based on repeat failures of the Division 2 emergency UPS, 2VBA*UPS2B. The review focused on: (1) proper maintenance rule (MR) scoping in accordance with 10 CFR 50.65; (2) characterization of failed structures, systems, and components (SSCs) safety significance classifications; and, (3) 10 CFR 50.65 (a)(1)/(a)(2) classification. The inspectors reviewed N2-MRM-REL-0104, "Maintenance Rule Scope," N2-MRM-REL-0105, "Maintenance Rule Performance Criteria," the third quarter 2004 System Health Report for Unit 2 DC Electric Power and UPS Systems, and DER NM-2004-3414, "2VBA*UPS2B has failed to the maintenance power supply."

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope (71111.13 - 5 Samples)

The inspectors reviewed five risk assessments and emergent work activities during this inspection period. For selected maintenance, work items or work orders the inspectors evaluated: (1) the effectiveness of the risk assessments performed before the maintenance activities were conducted; (2) risk management control activities; (3) the necessary steps taken to plan and control resultant emergent work tasks; and, (4) the overall adequacy of identification and resolution of emergent work and the associated maintenance risk assessments. GAP-OPS-117, "Integrated Risk Management," was used for this review. The following assessments/activities were reviewed:

- On January 5, power board 167 auto-transferred to the alternate power supply when a grounding clip in use for motor testing came in contact with energized conductors. The inspectors reviewed licensee compensatory actions for the unplanned entry into TS limiting conditions for operations (LCOs). Reference DER NM-2005-39 (Unit 1)
- On January 11, the air regulator for the 103 emergency diesel generator (EDG) starting air system was placed in service, causing the downstream relief valve to lift. The EDG was declared inoperable pending resolution of the starting air deficiency. The inspectors observed troubleshooting and repair activities during this unplanned entry into the TS LCO. Reference DER NM-2005-116 (Unit 1)
- On January 25, the Division 3 EDG was declared inoperable due to a problem with the lubricating oil system; although the normal AC-powered lube oil pump was running, the DC-powered pump was cycling on and off due to low system pressure. The inspectors reviewed the licensee's troubleshooting plan and corrective action implementation. Reference DERs NM-2005-348 and -359 (Unit 2)
- On February 3, the inspectors reviewed the impact of a failure of uninterruptible power supply 2VBB-UPS1B. Although this is a non-safety class UPS, it is classified under the Maintenance Rule program as high safety significant, and loss of its maintenance power supply would result in a plant trip. Reference DER NM-2005-461 (Unit 2)
- On March 9, the inspectors reviewed the risk assessment and compensatory measures for open circuiting the Division 2 battery while jumpering one cell due to low voltage. Reference DER NM-2005-811 (Unit 2)

b. Findings

Introduction. An NRC identified Green NCV of 10 CFR 50.65(a)(4) was identified for failure to adequately manage the increase in risk that resulted from maintenance on the Unit 2, Division 2, 125 Vdc battery (2BYS*BAT2B). Specifically, the fastener size was not adequately determined prior to installing a jumper around one of the battery cells, which resulted in the plant being maintained in a high risk configuration for approximately twice as long as would otherwise have been necessary.

Description. On February 28, during the performance of a weekly battery surveillance on 2BYS*BAT2B, cell 21 was found to have a voltage of 2.10 Vdc. TS Table 3.8.6-1, "Battery Cell Parameter Requirements," specifies a minimum cell voltage of 2.13 Vdc. TS 3.8.6.A requires that any cell not satisfying this requirement be restored to greater than 2.13 Vdc within 31 days. The battery was placed on equalizing charge for one week in an attempt to restore the voltage. In parallel, development of a temporary modification to jumper cell 21 was commenced.

The Temporary Change Package (TCP) TCP N2-05-027, "Jumper Cell 21 on Battery 2BYS*BAT2B," required that the battery breaker be open during performance of the maintenance. This constituted a high risk plant configuration, and the licensee performed a PRA-based risk analysis of the activity. The analysis indicated that the core damage frequency (CDF) while in this configuration was 4E-03/year. The NRC-accepted industry guideline, NUMARC 93-01, states that conditions with a configuration-specific CDF of greater than 1E-3/year should be carefully considered before voluntarily establishing such conditions, and then, only for very short periods of time. However, the same guideline also states that an activity with a conditional core damage probability (CCDP) of less than 1E-06 is not considered to be risk significant. By limiting the time that the battery breaker would be open to less than two hours, the CCDP would be 9E-07. Therefore, the licensee concluded that the activity would not pose an unacceptable risk.

The licensee implemented TCP N2-05-027 on March 9. Initial jumper installation was completed in approximately 36 minutes. However, it was noted that the bolts that had been used to attach the jumper terminals to the battery posts were too long and interfered with each other. This caused the jumper terminals to be unevenly forced apart, thereby potentially lessening contact with the battery posts. After discussion with Engineering, it was decided that the bolts should be removed and shortened from 1.5" to 1.25." This was done, and the resultant terminal configuration was satisfactory. However, disassembly, shortening the bolts, and reassembly had taken approximately 42 additional minutes to complete.

The inspectors reviewed TCP N2-05-027 and the implementing work order, 05-03362-00, to determine the cause of the unanticipated delay. The TCP stated that ½ " diameter by 1" long bolts may be used if full thread engagement is achieved, 1.25" or 1.5" length bolts shall be used if full thread engagement cannot be achieved with the 1" bolts. However, in development of the implementing work order, only 1" and 1.5" long bolts had been included on the list of parts, because it was known that 1.25" long bolts were not available in the warehouse. Rather than altering the TCP requirement based on unavailability, the 1.25" long bolts should either have been procured or fabricated ahead of time, or the actual required length should have been more precisely determined during development of the TCP. The inspectors concluded that the delay in completion of the jumpering activity constituted a violation of 10 CFR 50.65(a)(4), in that the licensee failed to adequately manage the increase in risk that resulted from the proposed maintenance activity by failing to have properly sized fasteners on hand prior to commencing the activity.

Analysis. The performance deficiency associated with this event was failure to adequately plan for jumpering battery 2BYS*BAT2B cell 21, such that the time spent in a high risk plant configuration would be minimized. The finding was greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance (Green) in accordance with Phase 1 of the Reactor Safety SDP because it was not a design or qualification deficiency that had been confirmed to result in a loss of function per Generic Letter 91-18, did not represent a loss of safety function, did not represent actual loss of safety function of a single train for greater than its TS allowed outage time, did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk-significant per 10 CFR 50.65 for greater than 24 hours, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

The failure to adequately manage the increase in risk that resulted from the battery maintenance was an example of a cross-cutting issue in human performance at the organizational level. Specifically, Engineering did not apply rigor commensurate with the sensitivity of the maintenance activity when they failed to determine the precise length of the required fasteners in developing the TCP; and, Maintenance personnel inappropriately excluded parts that were specified in the TCP when preparing for the activity, based on unavailability rather than technical justification.

Enforcement. 10 CFR 50.65(a)(4) states, in part, that, "Before performing maintenance activities . . . the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities . . ." Contrary to the above, on March 9, 2005, the licensee failed to adequately manage the increase in risk that resulted from the plant configuration that was established to support installation of a jumper around battery 2BYS*BAT2B cell 21, in that appropriately sized fasteners were not available prior to conducting the activity, which caused the amount of time that was required to complete the activity to be approximately doubled. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (DER NM-2005-1135), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000410/2005002-02, Failure to Adequately Manage Risk Associated with Maintenance to Jumper a Vital 125 VDC Battery Cell.

1R14 Operator Performance During Non-routine Evolutions and Events

a. Inspection Scope (71111.14 - 3 Samples)

On February 10, Unit 1 operators noted a prompt increase in drywell unidentified leakage. Investigation revealed that the cause was degradation of the 15 reactor recirculation pump (RRP) low pressure seal. The inspectors reported to the control room and observed the operators' response to this emergent condition. The licensee promptly developed limiting values for various pump and drywell parameters, at which the pump would be secured and isolated. The TS limit for drywell unidentified leakage was never exceeded, nor were any emergency operating procedure (EOP) entry conditions satisfied. After the initial increase in leakage, seal performance became relatively stable and, despite continuing gradual seal degradation, 15 RRP was able to be maintained in-service until shutdown for the refueling outage.

On March 7, at 4:37 a.m., Unit 1 experienced an automatic reactor scram. The cause of the scram was a turbine trip signal generated from a high level alarm in the 122 moisture separator. The inspectors responded to the control room upon arrival at the site, and observed licensee actions to control reactor cooldown rate and initiate shutdown cooling. The reactor was taken to cold shutdown conditions and repairs to the moisture separator level instrument were completed. A normal reactor startup was initiated on March 8, and the inspectors observed the approach to criticality and establishment of the initial heatup rate. The reactor reached 100 percent rated power on March 10.

On March 15, Unit 1 drywell pressure was observed to be lowering. The licensee took actions to identify any breaches of primary containment, including walkdowns of the RB and systems. The licensee added nitrogen to the drywell atmosphere to mitigate the pressure drop. The inspectors observed these efforts from the control room. No TS limits for drywell temperature or pressure were exceeded. The cause of the pressure drop was determined to be overcooling of the drywell atmosphere. While attempting to troubleshoot the reactor building closed loop cooling (RBCLC) heat exchanger tube leaks, cooling water flow to the drywell was increased. The increased flow through the drywell area coolers lowered the drywell temperature and pressure. RBCLC system alignment and flowrates were restored to nominal values, and drywell pressure and temperature recovered to normal.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15 - 7 Samples)

The inspectors reviewed seven operability evaluations during this inspection period, which affected risk significant mitigating systems, assessing: (1) the technical adequacy of the evaluation; (2) whether other existing degraded systems adversely impacted the affected system or compensatory measures; and, (3) where compensatory measures were used, whether the measures were appropriate and properly controlled; and, 4 that the degraded systems remained operable. Procedure S-ODP-OPS-0116, “Operability Determinations,” was used for this review. Operability evaluations associated with the following issues were reviewed:

- DER NM-2005-0318, Technical Support Center emergency ventilation system supply flow greater than surveillance acceptance criteria (Unit 1)
- DER NM-2005-0383, EDG 103 has delayed response to start initiation after inactivity of greater than seven days (Unit 1)
- DER NM-2005-0770, Degraded flow through the minimum flow line #12 control rod drive (CRD) pump (Unit 1)
- DER NM-2005-1021, Operability of primary containment after pressure decrease due to cooling water flow increase (Unit 1)
- DER NM-2005-370, Change in the leading edge flow meter correction factor reduced core thermal power by 18.7 megawatts thermal (Unit 2)
- DER NM-2005-346, E SW pump strainer element found broken off and missing in the SW system (Unit 2)
- DER NM-2005-811, Division 2 battery cell 21 voltage found to be less than the TS minimum (Unit 2)

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope (71111.19 - 4 Samples)

The inspectors reviewed post-maintenance testing (PMT) procedures and associated testing activities for four selected risk significant mitigating systems, assessing whether: (1) the effect of testing on the plant had been adequately addressed by control room and engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness, consistent with the design and licensing basis documents; (4) test instrumentation had current calibrations, and appropriate range and accuracy for the application; (5) tests were performed as written, with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; and, (7) test equipment was removed following testing and equipment was returned to the status required to perform its safety function. The following PMT activities were reviewed:

- N2-ESP-BYS-Q676, "Quarterly Battery Surveillance Test," Attachment 7, "Corrosion / Digital Low Resistance Ohmmeter (DLRO) Testing," after installation of a jumper around cell 21 in battery 2BYS*BAT2B (Unit 2)
- N1-ST-M4B, "Emergency Diesel Generator 103 and Power Board (PB) 103 Operability Test," after air start system maintenance (Unit 1)
- N1-ST-Q5, "Primary Containment Isolation Valve (IV) Operability Test," after environmental qualification (EQ) splice inspections on valve 05-05 (Unit 1)
- N1-ST-Q2, "Control Rod Drive Pumps Flow Rate Test," after #12 CRD pump motor replacement (Unit 1)

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22 - 7 Samples)

The inspectors witnessed performance of surveillance test procedures and/or reviewed test data of selected risk significant SSCs to assess whether the testing satisfied TS, FSAR/USAR, and licensee procedure requirements, and to determine if the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following surveillance tests were reviewed:

- N1-ISP-LRT-TYC, "Type "C" Containment Isolation Valve Leak Rate Test," for main steam isolation valve 01-01 (Unit 1)

- N1-ST M8, “Reactor Building Emergency Ventilation System Operability Test” (Unit 1)
- N!-ST-SO, “Shift Checks,” for Spent Fuel Pool liner leakage (Unit 1)
- N2-OSP-EGS-M@001, Diesel Generator and Diesel Air Start Valve Operability Test - Division II (Unit 2)
- N2-OSP-RHS-Q@006, RHR System Loop C Pump and Valve Operability Test and System Integrity Test (Unit 2)
- N2-OSP-ISC-M@002, Drywell Vacuum Breaker Operability Test (Unit 2)
- N2-OSP-EGS-M@002, Diesel Generator and Diesel Air Start Valve Operability Test - Division III (Unit 2)
- N1-ST-V19, Emergency Cooling System - Heat Removal Capability Test at High Power (Sample previously in Inspection Report 05000220/2004002)

b. Findings

Introduction. A Green NCV of Unit 1 TS 6.4.1.a was identified concerning an inadequate procedure review and approval process related to the development of procedure N1-ST-V19, “Emergency Cooling System - Heat Removal Capability Test at High Power.” Specifically, applicability of the 10 CFR 50.59 screening process was incorrect as the licensee had inappropriately determined that all aspects of the activity were controlled by other processes, thereby negating the requirement for a 10 CFR 50.59 screen. Subsequently it was determined that the procedure also contained changes that affect operation and control of other systems and therefore that a 10 CFR 50.59 screen should have been completed.

Description. In 2003, a new surveillance test procedure, N1-ST-V19, was developed to perform emergency condenser (EC) heat exchanger capacity testing at higher reactor power than had previously been done. The test procedure was first performed in January 2004 while shutting down for a planned maintenance outage. The test was aborted due to a greater than expected transient increase of reactor thermal power rise of more than 20 MWt. NRC Inspection Report 05000220/2004002 documented that the procedure had not provided operators with comprehensive and appropriate limitations concerning reactor response upon initiation of the emergency condenser system. A Green NCV was documented for an inadequate procedure for performing the EC system heat exchanger capacity test. In that NRC inspection report, the procedure development aspects of the issue were not addressed.

The licensee’s procedure development is governed by NIP-PRO-03, “Preparation and Review of Technical Procedures.” The procedure states that procedure preparation should be consistent with the requirements of related committed documents, including 10 CFR 50.59 evaluations and also that technical verifiers shall ensure that the procedure is consistent with reference documents established as requirements including 10 CFR 50.59. NIP-DSE-01, “10 CFR 50.59 Applicability Determinations, Screens and Evaluations,” accomplishes this function. Using that procedure’s 10 CFR 50.59 applicability determination process, the licensee improperly determined that a 10 CFR 50.59 screen was not required. The basis for the decision was that the test procedure constituted a maintenance activity that was controlled by alternative processes including

10CFR50.65(a)(4) and 10CFR50, Appendix B Quality Assurance program controls for procedure development.

Subsequently, inspectors determined that N1-ST-V19 also contained changes that affect operation and control of other systems and therefore that a 10 CFR 50.59 screen should have been initiated. On March 18, 2005, the licensee completed a 10 CFR 50.59 screening per NIP-DSE-01 and determined that the EC system heat removal capacity test procedure operates the reactor pressure and level control systems consistent with the USAR and that the reactor's operation and control were not impacted by the surveillance test.

Analysis. Notwithstanding the acceptable result of the 10 CFR 50.59 screen, the performance deficiency associated with this event was a failure to perform a 10 CFR 50.59 screen. The procedure development process should ensure that 10 CFR 50.59 is applied such that maintenance procedure changes do not inadvertently alter the operation or control of SSCs.

The performance deficiency is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affected the associated cornerstone objective of ensuring the capability of the emergency condenser system, a core decay heat removal system, to respond to initiating events to prevent undesirable consequences. Using Phase 1 of the Reactor Safety SDP the finding was determined to be of very low safety significance (Green) because it was not a design or qualification deficiency and it did not represent an actual loss of the emergency condenser system safety function, and was not potentially risk significant due to seismic, flood, fire or weather related initiating events.

Enforcement. Unit 1 TS 6.4.1a requires, in part, that written procedures shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7 - 1972 and cover applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 3, 1972. Regulatory Guide 1.33, Appendix A, November 3, 1972, requires procedure review and approval processes be covered by written procedures. NIP-PRO-03, "Preparation and Review of Technical Procedures," states that procedure preparation should be consistent with the requirements of related committed documents including 10 CFR 50.59 evaluations. N1-ST-V19, "Emergency Cooling System - Heat Removal Capability Test at High Power," was prepared using this process.

Contrary to the above on January 9, 2004, the preparation and review of the heat removal capacity test was inadequate in that it was not consistent with the requirements of the related committed document, 10 CFR 50.59, in that a 10 CFR 50.59 screen was not performed. However, because of the very low safety significance and because the corrective actions taken through DER NM-2005-539 appeared to be reasonable, the issue is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000220/2005002-03, Failure to Perform a 50.59 Screen During Development of an Emergency Condenser Capacity Test.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23 - 3 Samples)

The inspectors reviewed the following temporary plant modifications to determine whether the temporary change adversely affected system or support system availability, or adversely affected a function important to plant safety. The inspectors reviewed the associated system design bases, including the FSAR and TS, and assessed the adequacy of the safety determination screening and evaluation. The inspectors also assessed configuration control of the temporary change by reviewing selected drawings and procedures to verify whether appropriate updates had been made. The inspectors compared the actual installation with the temporary modification documents to determine whether the implemented change was consistent with the approved documented modification. The inspectors reviewed the post-installation test results to verify whether the actual impact of the temporary change had been adequately demonstrated by the test.

- The inspectors reviewed Unit 2 Temporary Change N2-04-171, “Install Temporary Fiber Optic Patch Cables for Average Power Range Monitor (APRM) 2.” This modification temporarily replaces damaged fiber optic cables between APRM 2 and local power range monitor (LPRM) 2, and rod block monitor (RBM) B, with fiber optic patch cables. The cables provide for digital communications between these instruments. This modification will be removed during the next refueling outage, when replacement original cables will be installed.
- The inspectors reviewed Unit 2 Temporary Change N2-05-027, “Jumper Cell 21 on Battery 2BYS*BAT2B.” The condition that led to the development and implementation of this temporary modification are discussed in section 1R13 of this report.
- The inspectors reviewed Unit 1 Temporary Change N1-04-165, “SFC Temporary Cooling System for RFO18,” due to the increased importance of fuel pool cooling while the reactor core was offloaded during the refueling outage.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope (71114.04 - 1 Sample)

An in-office inspection that reviewed recent changes to the emergency plan procedures was conducted on March 8. The review verified the changes, satisfied the standards of 10 CFR 50.54(q), 10 CFR 50.47(b), the requirements of 10 CFR 50 Appendix E, the intent of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," and that the changes did not decrease the effectiveness of the plan. These changes are subject to future NRC inspections to ensure that as a result of these changes the emergency plan continues to meet NRC regulations.

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 (71121.01 - 9 Samples)

The inspector identified exposure significant work areas within radiation areas, high radiation areas (<1 R/hr), or airborne radioactivity areas in the plant and reviewed associated licensee controls and surveys of these areas to determine if the controls (e.g. surveys, postings, barricades) were acceptable.

The inspector walked down these areas or their perimeters to determine: whether prescribed radiation work permit (RWP), procedure, and engineering controls were in place; whether licensee surveys and postings were complete and accurate; and, whether air samplers were properly located.

The inspector reviewed RWPs used to access these and other high radiation areas and identify what work control instructions or control barriers had been specified and reviewed electronic personal dosimeter (EPD) alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy.

Based on the licensee’s schedule of work activities, the inspector selected three jobs being performed in radiation areas, airborne radioactivity areas, or high radiation areas (<1 R/hr) for observation (drywell chemical decontamination, shutdown cooling valve 38-13 removal, and control rod drive work); reviewed radiological job requirements (RWP requirements and work procedure requirements); observed job performance with respect to these requirements; and, determined that radiological conditions in the work area were adequately communicated to workers through briefings and postings.

During job performance observations, the inspector verified the adequacy of radiological controls, such as: required surveys (including system breach radiation, contamination, and airborne surveys), radiation protection job coverage(including audio and visual surveillance for remote job coverage), and contamination controls.

For high radiation work areas with significant dose rate gradients (factor of 5 or more), the inspector reviewed the application of dosimetry to effectively monitor exposure to personnel and verified that licensee controls were adequate.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope (71121.02 - 3 Samples)

Based on scheduled work activities and associated exposure estimates, the inspector selected three work activities listed in paragraph 2OS1 above, in radiation areas, airborne radioactivity areas, or high radiation areas for observation. The inspector evaluated the licensee's use of as low as is reasonably achievable (ALARA) controls for these work activities by performing the following: evaluated the licensee's use of engineering controls to achieve dose reductions; evaluated procedures and controls to verify consistency with the licensee's ALARA reviews; verified sufficient shielding of radiation sources provided for; verified dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope (71121.03 - 1 Sample)

The inspector conducted a review of selected radiation protection instruments located in the radiologically controlled area (RCA). Items reviewed were: verification of proper function; certification of appropriate source checks; and calibration for those instruments used to ensure that occupational exposures were maintained in accordance with 10 CFR 20.1201.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

1. Corrective Action Review by Resident Inspectors

a. Inspection Scope(71152)

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 a daily screening of items entered into the Nine Mile Point corrective action program. This review was accomplished by reviewing paper copies of each condition report, attending daily screening meetings, and accessing Constellation Energy’s computerized database.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

1. (Closed) LER 05000220/2004-002-00, Changes and Errors in the Methodology Used by General Electric and Global Nuclear Fuel to Demonstrate Compliance with Emergency Core Cooling System Performance.

On May 14, 2004, the licensee received notification from General Electric of a change in the calculation of peak cladding temperature and maximum local cladding oxidation. A new heat source has been postulated during the loss of coolant accident (LOCA) which involved the recombination of hydrogen and oxygen within the fuel bundles during core heatup. Consequently, the previous LOCA analysis was potentially non-conservative. The licensee determined that the cause of this event was that the heating effects of the hydrogen-oxygen recombination phenomenon were not properly considered during the original development of the LOCA evaluation methodology. Corrective actions included reducing the maximum average planar linear heat generation rate thermal limit and further analyses to evaluate the hydrogen-oxygen recombination phenomenon. The inspectors reviewed this licensee event report (LER) and no findings of significance were identified since it was not a result of a licensee performance deficiency and therefore not evaluated as a potential finding. This LER is closed.

2. (Closed) LER 05000220/2004-003-00, Inadequate Environmental Qualification Barrier Considerations Resulting in an Unanalyzed Condition.

Introduction. An NRC identified Green NCV of 10 CFR 50.65(a)(4) was identified for failure to adequately assess the increase in risk that resulted from maintenance on the Unit 1 control room ventilation system. Specifically, no assessment of risk was performed prior to opening doors which served as barriers between the mild environment of the Control Room and the potential harsh environment of the Turbine Building resulting from a high energy line break (HELB).

Description. On February 25, 2003, with Unit 1 operating at 100 percent power, several doors to the Unit 1 Control Room and Auxiliary Control Room were opened for ventilation purposes while the control room ventilation system was out-of-service for maintenance. Two of the doors were part of the high energy line break boundary, between the Control Room and the Turbine Building. The inspectors questioned the environmental qualification (EQ) implications on control room equipment of the open doors. The doors were subsequently closed which restored the control room EQ boundary. The breach permits which allowed opening the doors during the maintenance were reviewed and had been issued in accordance with Unit 1 procedures. The procedure for breech permits did not include EQ considerations as an attribute to be considered when processing breach permits. The inspectors concluded that the failure to account for the impact on environmental qualification of control room equipment constituted a violation of 10CFR 50.65(a)(4), in that the licensee failed to assess the risk that resulted from the maintenance activity.

Analysis. The performance deficiency associated with this event was failure to adequately assess the increased risk from a HELB in the Turbine Building with doors in the HELB boundary open to the Control Room. The finding was more than minor because of left uncorrected, it would become a more significant safety concern in that actions to assess and manage increases in risk may not have been implemented. Using the Phase 1 worksheets in Manual Chapter 0609, “Significance Determination Process,” the inspectors determined that the finding affected the Mitigating Systems and Barrier Integrity Cornerstones. As a result, the Region I Senior Reactor Analyst (SRA) conducted a Phase 3 analysis and determined that the performance deficiency was of very low safety significance (Green), with an increase in CDF in the mid-E-8 range (approximately one additional core damage event in 20,000,000 years of reactor operation). The risk determination assumed that the doors were open for 53 hours, with a HELB (main steam or feed line break in the Turbine Building) frequency of one in 1000 years of reactor operation, and a 1 in 100 chance that the main steam isolation valves or feed line check valves would not close.

Enforcement. 10 CFR 50.65(a)(4) states, in part, that, “Before performing maintenance activities . . . the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities . . .” Contrary to the above, on February 25, 2003, the licensee failed to adequately assess risk that resulted from opening HELB barriers to support maintenance activities on the control room ventilation system. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (DER NM-2003-1499), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000220/2005002-04, Failure to Adequately Assess Risk Associated with Maintenance on the Control Room Ventilation System. This LER is closed.

3. (Closed) LER 05000220/2004-004-00, Manual Reactor Scram Due to Failure of #13 Feedwater Flow Control Valve Positioner.

On August 30, 2004, Unit 1 experienced feedwater flow oscillations. Operator efforts to manually stabilize feedwater flow were unsuccessful and a manual scram was initiated to shutdown the plant. All systems worked as designed to stabilize the reactor (See NRC Inspection Report 05000220/2004004, Section 1R14).

The cause of the flow oscillation was determined to be the failure of a diaphragm in the pneumatic module for the feedwater flow control valve positioner. The failed positioner had been in service since the fall of 2000, approximately four years. The diaphragm was operated at the maximum vendor recommended air supply pressure, which combined with the duty cycle, resulted in reduced service life. The licensee determined the root cause to be that the original design did not adequately establish service life given the operating conditions. Licensee corrective action was to change out the positioner on two year frequency based on diaphragm service life. The inspectors reviewed this LER and no findings of significance were identified. This LER is closed.

4OA4 Cross-Cutting Aspects of Findings

Section 1R13 describes a cross-cutting issue in the area of human performance at the organizational level. Specifically, the Engineering and Maintenance Departments failed to adequately manage the increase in risk that resulted from maintenance on the Unit 2, Division 2, 125 VDC battery (2BYS*BAT2B). Specifically, Engineering did not apply rigor commensurate with the sensitivity of the maintenance activity when they failed to determine the precise length of the required fasteners in developing the TCP; and, Maintenance personnel inappropriately excluded parts that were specified in the TCP when preparing for the activity, based on unavailability rather than technical justification.

4OA6 Meetings, Including Exit

On April 15, 2005, the inspectors presented the inspection results to Mr. Tim O'Connor, and other members of licensee management. The licensee confirmed that proprietary information was not provided during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

J. Gerber, ALARA Supervisor
R. Godley, Manager, Operations
B. Holston, Manager, Engineering Services
A. Julka, CEG, Director, Q&PA
T. Kulczycky, Reliability Engineering
S. Leonard, CEG, GS Licensing
T. O'Connor, Plant General Manager
W. Paulhardt, Manager, Radiation Protection
G. Perkins, General Supervisor, Engineering Programs
J. Spina, Site Vice President
T. Syrell, Nuclear Regulatory Matters

NRC Personnel

W. Schmidt, Sr. Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000410/2005002-01	FIN	Improper Installation of HPCS Suction Line Flexible Coupling Due To Inadequate Procedure
05000410/2005002-02	NCV	Failure to Adequately Manage Risk Associated with Maintenance to Jumper a Vital 125 VDC Battery Cell
05000220/2005002-03	NCV	Failure to Perform a 50.59 Screen During Development of an Emergency Condenser Capacity Test
05000220/2005002-04	NCV	Failure to Adequately Assess Risk Associated with Maintenance on the Control Room Ventilation System

Closed

05000220/2004-002-00	LER	Changes and Errors in the Methodology Used by General Electric and Global Nuclear Fuel to Demonstrate Compliance with Emergency Core Cooling System Performance

05000220/2004-003-00	LER	Inadequate Environmental Qualification Barrier Considerations Resulting in an Analyzed Condition
05000220/2004-004-00	LER	Manual Reactor Scram Due to Failure of #13 Feedwater Flow Control Valve Positioner

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

DER NM-2005-54

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

DER NM-2005-811
NUMARC 93-01, Revision 3, Section 11.0, Assessment of Risk Resulting From Performance of Maintenance Activities

Section 1R22: Surveillance Testing

DER NM-2005-539
NEI 96-07, Revision 1, Guidelines for 10 CFR 50.59 Implementation

Section 1EP4: Emergency Plan and Implementing Procedures Changes

EPIP-EPP-01	Classification of Emergency Conditions at Unit 1, Revision 13
EPIP-EPP-02	Classification of Emergency Conditions at Unit 2, Revision 13
EPIP-EPP-03	Search and Rescue, Revision 05
EPIP-EPP-04	Personnel Injury or Illness, Revision 10
EPIP-EPP-05B	Protected Area Evacuation, Revision 02
EPIP-EPP-05C	Exclusion Area Evacuation, Revision 02
EPIP-EPP-05D	Accountability, Revision 02
EPIP-EPP-10	Security Contingency Event, Revision 08
EPIP-EPP-11	Hazardous Material Incident Response, Revision 08
EPIP-EPP-18	Activation and Direction of Emergency Plans, Revision 13
EPIP-EPP-23	Emergency Personnel Action Procedures, Revision 17
EPIP-EPP-31	Control Room Support Functions from the TSC, Revision 02
EPMP-EPP-0101	Unit 1 Emergency Classification Technical Bases
EPMP-EPP-0102	Unit 2 Emergency Classification Technical Bases

<u>Section 20S2: ALARA Planning and Controls</u>

<u>ALARA Reviews</u>:

05-1-02	Drywell 259' Permanent Shielding Modification
05-1-03	Chemical Decontamination
05-1-09	Replace Bottom Head Drain Line Valve VLV-37-10
05-1-12	Control Rod Drive Work
05-1-15	Drywell Scaffold
05-1-16	Drywell Insulation
05-1-18	Drywell Motor Operated Valves
05-1-19	Replace Shutdown Cooling Valve IV-38-13
05-1-20	Reactor Refueling and Inspection
05-1-24	Drywell In-Service Inspection

LIST OF ACRONYMS

ADAMS	agencywide documents access and management system
ALARA	as low as is reasonably achievable
APRM	average power range monitor
CCDP	conditional core damage probability
CDF	core damage frequency
CFR	Code of Federal Regulations
CRD	control rod drive
CST	condensate storage tank
DER	deviation/event report
DLRO	digital low resistance ohmmeter
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOP	emergency operating procedure
EPD	electronic pocket dosimeter
EQ	environmental qualification
ERO	emergency response organization
FIN	finding
FSAR	final safety analysis report
HELB	high energy line break
HPCS	high pressure core spray
IMC	inspection manual chapter
IPE	individual plant examination
IR	inspection report
IV	isolation valve
LCO	limiting condition for operation
LER	licensee event report
LOCA	loss of coolant accident
LPRM	local power range monitor
MC	manual chapter
MR	maintenance rule
NCV	non-cited violation
NMPNS	Nine Mile Point Nuclear Station

NRC	U.S. Nuclear Regulatory Commission
PARS	publically available records
PB	power board
PMT	post-maintenance testing
PRA	probabilistic risk assessment
RB	reactor building
RBCLC	reactor building closed loop cooling
RBM	rod block monitor
RCA	radiologically controlled area
RFO	refueling outage
RHR	residual heat removal
RRP	reactor recirculation pump
RWP	radiation work permit
SDP	significance determination process
SFC	spent fuel cooling
SRA	senior reactor analyst
SSCs	structures, systems, and components
SW	service water
TCP	temporary change package
TS	technical specification
UPS	uninterruptible power supply
USAR	updated safety analysis report
VDC	volts direct current