October 29, 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 05000220/2005004 and 05000410/2005004

Dear Mr. Spina:

On September 30, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Nine Mile Point Nuclear Station (NMPNS) Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on October 14, 2005, with Mr. Tim O'Connor and other members of your staff.

The 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 two NRC-identified findings of very low safety significance (Green). The findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these two violations as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, 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, 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

NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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 05000220/2005004 and 05000410/2005004
w/Attachment: Supplemental Information

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REGION I

Docket Nos.:	50-220, 50-410
License Nos.:	DPR-63, NPF-69
Report No.:	05000220/2005004 and 05000410/2005004
Licensee:	Nine Mile Point Nuclear Station, LLC (NMPNS)
Facility:	Nine Mile Point, Units 1 and 2
Location:	Lake Road Oswego, NY
Dates:	July 1, 2005 through September 30, 2005
Inspectors:	G. Hunegs, Senior Resident Inspector B. Fuller, Resident Inspector E. Knutson, Resident Inspector F. Arner, Senior Reactor Inspector P. Finney, Reactor Inspector J. Furia, Senior Health Physicist K. Mangan, Senior Reactor Inspector
Approved by:	James M. Trapp, Chief Projects Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000220/2005-004, 05000410/2005-004; 07/01/05 - 09/30/05; Nine Mile Point, Units 1 and 2; Operator Performance During Non-routine Evolutions and Events, and Operability Evaluations.

This report covered a 3-month period of inspection by resident inspectors, and two announced inspections by four region-based inspectors. Two Green findings, all of which were non-cited violations (NCVs), 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: Initiating Events, Mitigating Systems

- Green. The inspectors identified an NCV of 10 CFR 50.65(a)(4) for the failure to assess and manage the increase in risk associated with power board maintenance which resulted in an unplanned reactor scram. The performance deficiency associated with this event was the failure to assess and manage the risk and recognize the plant impact associated with power board 11 breaker maintenance coincident with reactor protection system testing on the other channel. A contributing cause of the finding is related to the cross-cutting element of human performance.

  The finding is greater than minor because it is associated with the Initiating Events Cornerstone attribute of human performance and affects the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. In addition, the finding relates to the maintenance risk assessment and risk management issue of the failure of the licensee's risk assessment to consider maintenance activities that could increase the likelihood of initiating events. The finding is also associated with the Mitigating Systems Cornerstone because the loss of power board 11 caused a loss of one train of feedwater coolant injection. The finding was determined to be of very low safety significance in accordance with Phase 3 of the Reactor Safety SDP. (Section 1R14)

Cornerstone: Barrier Integrity

- Green. The inspectors identified an NCV of 10 CFR 50 Appendix B, Criterion XI, "Test Control," for failure to conduct testing to determine torus-to-drywell vacuum relief check valve's opening force under actual in-service conditions. Specifically, four vacuum relief check valves were being cycled open and closed prior to measurement of their opening force. The performance deficiency associated with this issue is an inadequate surveillance procedure, in that the licensee failed to recognize that cycling all of the torus-to-drywell vacuum relief valves prior to measurement of their opening force was unnecessary and constituted unacceptable preconditioning. (i.e. potentially altering the amount of force required to open the valves during the actual test)

The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of the procedure quality of a risk important surveillance and affects the cornerstone objective of providing reasonable assurance that physical design barriers (specifically, the primary containment) protect the public from radionuclide releases caused by accidents or events. The finding is determined to be of very low safety significance (Green) in accordance with Phase I of the Reactor Safety Significance Determination Process (SDP) because it did not represent a degradation of the radiological barrier function provided for the control room, spent fuel pool, or standby gas treatment system, did not represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere, and did not represent an actual open pathway in the physical integrity of reactor containment or involve an actual reduction in defense-in-depth for the atmospheric pressure control or hydrogen control functions of the reactor containment. (Section 1R15)

B. <u>Licensee-Identified Violations</u>

None

REPORT DETAILS

Summary of Plant Status

Nine Mile Point Unit 1 (Unit 1) began the inspection period at 100 percent power. On August 18, an automatic scram occurred due to a loss of power board 11 coincident with a half scram already present on the reactor protection system (RPS) channel 12 due to instrument and control testing. The loss of power board 11 caused a loss of 11 reactor protection system trip bus, which in turn produced a half scram. The reactor was started up on August 19 and returned to service on August 20. Unit 1 reached 100 percent power on August 20 and operated there through the end of the inspection period.

Nine Mile Point Unit 2 (Unit 2) began the inspection period at 100 percent power. On September 9, power was reduced to 67 percent for control rod sequence exchange, scram time testing and channel bow testing. Power was returned to 100 percent on September 11 and Unit 2 operated there through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 Sample)

a. Inspection Scope

The inspectors performed one review of Unit 1 preparations for expected severe weather. On September 29, the inspectors reviewed Unit 1 contingency plans in preparation for potential severe thunderstorms. Unit 1 documents reviewed included N1-OP-64, "Meteorological Monitoring," and EPIP-EPP-26, "Natural Hazard Preparation and Recovery."

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests, or Experiments (71111.02 - 21 Samples)

a. Inspection Scope

The inspectors reviewed a sample of six safety evaluations among the initiating events, barrier integrity, and mitigating systems cornerstones to verify that changes and tests were reviewed and documented in accordance with 10 CFR 50.59 and when required, NRC approval was obtained prior to implementation. The inspectors assessed the adequacy of the safety evaluations through interviews with the plant staff and review of supporting information, such as calculations and analyses, design change documentation, procedures, the Updated Final Safety Analysis Report (UFSAR), technical specifications (TSs) and plant drawings. In addition, the inspectors reviewed the administrative procedures that control the screening, preparation, and issuance of the safety evaluations to ensure that the procedures adequately implemented the requirements of 10 CFR 50.59, "Changes, Tests, and Experiments."

The inspectors also reviewed a sample of fifteen changes that the licensee had evaluated (using a screening process) and determined to be outside of the scope of 10 CFR 50.59. The inspectors performed this review to assess if licensee conclusions with respect to 10 CFR 50.59 applicability were appropriate. The sample of issues that were screened out included design changes, procedure changes, UFSAR changes, setpoint changes, and calculation revisions. A listing of the safety evaluations, 50.59 screens and other documents reviewed is provided in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial Walkdown (71111.04 - 3 Samples)

The inspectors performed a partial walkdown of the following three systems 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 July 26, the inspector performed a partial walkdown of the Unit 2 A residual heat removal (RHR) subsystem, based on safety significance. Procedure N2-OP-31, "RHR System," was used for this review.
- On August 25, the inspectors performed a partial walkdown of the Unit 2 reactor core isolation cooling system due to increased risk when the high pressure core spray (HPCS) system was out of service. Procedure N2-OP-35, "Reactor Core Isolation Cooling," was used for this review.
- On August 25, the inspectors performed a partial walkdown of Unit 1 instrument air system when the 13 instrument air compressor was out of service. Procedure N1-OP-20, "Service, Instrument Air and Breathing Systems," was used for this review.

Complete Walkdown (71111.04S - 1 Sample)

The inspectors performed a complete walkdown of the Unit 2 standby gas treatment system. The walkdown was conducted to identify any discrepancies between the existing equipment alignment and the required alignment. The inspectors determined the correct system lineup using procedures N2-OP-61B, "Standby Gas Treatment System," and N2-VLU-01, "Walkdown Order Valve Lineup and Valve Operations," Attachment 61B, "N2-OP-61B Walkdown Valve Lineup," along with the appropriate piping and instrument drawings. In addition, the inspectors reviewed maintenance rule status, operator workarounds, and outstanding maintenance work requests and historical deficiencies that could potentially affect the ability of the system to perform its design basis function and to assess overall system health. During this inspection the inspectors verified the following: valves were correctly positioned; electrical power was available as required; labeling was correct; and valves required to be locked were properly locked. Minor issues identified were provided to system engineering personnel.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Protection - Tours (71111.05 Q - 7 samples)

a. Inspection Scope

The inspectors walked down accessible portions of the seven 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 USAR.

- Unit 1 Reactor Building (RB) 340 ft
- Unit 1 RB 318 ft
- Unit 1 Turbine Building (TB) 261 ft
- Unit 2 Division 1 switchgear room
- Unit 2 Division 2 switchgear room
- Unit 2 Division 1 EDG room
- Unit 2 Division 2 EDG room

b. Findings

No findings of significance were identified.

.2 Fire Protection - Drill Observation (71111.05A - 1 Sample)

a. Inspection Scope

The inspectors observed a fire brigade drill conducted on September 13, involving a simulated fire in the Unit 2 RB, and including participation by the Town of Scriba fire department. The inspectors evaluated the fire brigade response to and conduct of fire fighting activities by observing the following aspects: protective clothing properly donned; self-contained breathing apparatus properly worn; fire hose lines properly laid out and capable of reaching all necessary fire hazard locations; fire area of concern approached in a controlled manner; sufficient firefighting equipment brought to the scene; fire brigade leaders' directions thorough, clear, and effective; and effective coordination with the off-site fire department. The fire brigade drill was declared a "failed drill" as a result of several scenario objectives that were not met. The objectives that were not met were related to communication issues, command and tactical issues, and offsite interface issues. The licensee disqualified the fire brigade, pending remediation, consistent with procedure requirements. The inspector reviewed NTP-TQS-402, "Fire Brigade Drill Assessment," and DERs NM-2005-3534 and 3614 which documented the observed fire brigade response. The inspectors observed no violation of NRC requirements.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q - 1 Sample and 71114.06 - 1 Sample)

a. Inspection Scope

The inspectors observed Unit 1 licensed operator simulator training on July 27, to assess the licensee's training program effectiveness. 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 the emergency response organization (ERO) performance regarding initial and subsequent actions by licensed operators.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 2 Samples)

a. Inspection Scope

The inspectors reviewed one performance-based problem during this inspection period, involving the Unit 1 Feedwater level control system, to assess the effectiveness of the maintenance program. In addition, the inspectors reviewed the performance and condition history of one high safety significant system, the Unit 2 emergency diesel generators. Reviews 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) classifications. The inspectors reviewed the Unit 1 FSAR and Unit 2 USAR, procedures N1 and N2-MRM-REL-0104, "Maintenance Rule Scope," and N1 and

N2-MRM-REL-0105, “Maintenance Rule Performance Criteria,” and the applicable system health reports.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 6 Samples)

a. Inspection Scope

The inspectors reviewed six 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 July 16, the inspectors evaluated licensee activities associated with reactor vessel level variations caused by signal spiking from the 11 feedwater (FW) level column. Reference DER NM-2005-2793 (Unit 1).
- On July 20, the inspectors evaluated licensee activities associated with the unplanned LCO entry after momentary loss of 115KV line 4. Reference DER NM-2005-2812 (Unit 1).
- On July 25, the inspectors evaluated licensee scheduling activities that resulted in a Yellow PRA risk due to weekly half scram testing coincident with diesel generator testing (Unit 1).
- On August 18, the inspectors evaluated licensee activities associated with performance of WO-05-1580, N1-EPM-GEN 150, “Breaker Inspection and Preventive Maintenance” for R110, feeder to heater board 11H. Reference DER NM-2005-3175 (Unit 1).
- On August 25, the inspectors observed emergent maintenance to replace the FW computer module used in three element control of reactor vessel water level (Unit 1).
- On August 31, the inspectors reviewed the risk assessment for a two day planned maintenance outage for the Division 2 standby liquid control system, which resulted in increased risk during several other concurrently planned activities (Unit 2).

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-routine Evolutions and Events (71111.14 - 1 Sample)

a. Inspection Scope

On August 18, Unit 1 scrammed from 100 percent power due to a loss of power board 11 coincident with a half scram present already on reactor protection system channel 12 due to instrument and control testing.

The inspectors responded to the control room to monitor operator and plant response. The inspectors reviewed operator logs, plant computer data, charts, and DERs to determine what occurred, how the operators responded, and whether the response was in accordance with plant procedures. The inspectors reviewed event notification 41927 and N1-REP-6, "Post -Scram Review."

b. Findings

Introduction. An NRC-identified Green NCV of 10 CFR 50.65(a)(4) was identified for the failure to assess and manage the increase in risk associated with power board maintenance, coincident with reactor protection system testing on the other channel. Specifically, while a breaker was being installed in the power board, relays were inadvertently actuated which resulted in the loss of the power board and caused an unplanned reactor scram.

Description. On August 18, Unit 1 scrammed from 100 percent power due to a loss of 4.16 kv power board 11 (the power supply for RPS bus 11) coincident with a half scram present already on reactor protection system (RPS) channel 12 due to planned instrument and control testing in accordance with N1-IPM-032-008T, "Reactor Recirculation Flow Loop Calibration." The plant responded as designed and was stabilized with reactor level and pressure in the normal control bands.

A loss of power board 11 caused a loss of 11 RPS trip bus which resulted in a half scram. Reactor recirculation pumps 11 and 12 tripped due to loss of power board 11. The feedwater coolant injection (FWCI) system initiated due to the turbine trip. The 11 feedwater pump was in service until the loss of power board 11 resulted in the pump trip and the 12 feedwater pump autostarted on the FWCI low reactor vessel water level initiation signal. The cause of the loss of power board 11 was determined to be that, during installation of a non safety-related (auxiliary electric boiler) feeder breaker that was being installed into the breaker cubicle, a relay was inadvertently actuated which resulted in 11 power board tripping.

The licensee performed a root cause evaluation of the event and determined several work planning deficiencies related to risk assessment process integration into work planning and execution. The operators' plant impact review did not include an assessment of potential interactions with surrounding equipment as a result of the maintenance. It was not identified by individuals involved in the breaker maintenance or work planning that the maintenance activity could result in relay actuation which could result in loss of the power board.

Analysis. The performance deficiency associated with this event is the failure to assess and manage the risk and recognize the plant impact associated with power board 11 breaker maintenance being conducted coincident with reactor protection system testing. The finding is greater than minor because it is associated with the Initiating Events Cornerstone attribute of human performance and affects the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. In addition, the finding relates to the maintenance risk assessment and risk management issue of the failure of the licensee's risk assessment's to consider maintenance activities that could increase the likelihood of initiating events. The finding is also associated with the Mitigating Systems Cornerstone because the loss of power board 11 caused a loss of the 11 FWCI train.

In accordance with MC 0609 the inspectors conducted a significance determination process Phase 1 screening and determined that this finding required a Phase 2 assessment because two cornerstones were affected. Due to limitations of Phase 2 notebooks, a Senior Reactor Analyst (SRA) was obligated to conduct a Phase 3 risk assessment. The Phase 3 risk assessment was conducted using the Nine Mile Point Unit 1 Standardized Plant Analysis Risk (SPAR) model, revision 3.20 and SAPHIRE/GEM Version 7.0, revision 26, and determined that this performance deficiency was of very low risk significance (Green). The SRA performed a conditional risk assessment with the following assumptions: failure probability of power board No. 11 set to True (1.0), with no operator recovery action credit provided; the annualized transient initiating event frequency was conservatively increased by a factor of two (1.6 vice .8 events/year) based upon the direct impact of the performance deficiency; and an estimated exposure period of 4 hours was used. The approximate change in core damage frequency as a result of this performance deficiency was conservatively calculated by the SPAR model to be in the mid E-11 per year range, or very low risk significance. The most dominant core damage sequence involved a transient initiator (reactor trip) with: loss of the power conversion system; loss of isolation condensers; loss of high pressure makeup; and failure of operators to depressurize the reactor. For this specific finding, emergency AC buses were always available and no emergency core cooling systems were unavailable due to test or maintenance. In addition, the short duration of the performance deficiency greatly reduced the overall risk contribution to this event. The inspectors also determined that a contributing cause of this finding was related to the human performance cross-cutting area.

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 August 18, 2005, the licensee failed to adequately assess and manage the increase in risk during maintenance on a breaker located on power board 11. The breaker maintenance resulted in the loss of power board 11, (the power supply for RPS bus 11) which, coincident with reactor protection system testing on the other channel, resulted in a reactor scram. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (DER NM-2005-3175), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC

Enforcement Policy: NCV 05000220/2005004-01, Failure to Manage Risk Associated With Maintenance on Power Board 11 Breaker Resulted in a Reactor Scram.

1R15 Operability Evaluations (71111.15 - 4 Samples)

a. Inspection Scope

The inspectors reviewed four 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; (3) where compensatory measures were used, whether the measures were appropriate and properly controlled; and (4) whether 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:

- 12 reactor FW pump reactor building closed loop cooling (RBCLC) heat exchanger through wall leak. Reference DER NM-2005-2736.
- Division 3 EDG day tank draining back to the fuel oil storage tank. Reference DER NM-2005-2673.
- Service water (SW) strainer element was found broken off of the F-SW strainer. Reference DER NM-2005-2988.
- Torus to reactor building vacuum relief valve 68-05 operated abnormally following limit switch replacement (hesitated and then popped open). The inspectors examined the basis for subsequently determining it to be operable. Reference DER NM-2005-3056.

b. Findings

Introduction. An NRC identified Green NCV of 10 CFR 50 Appendix B, Criterion XI, “Test Control,” was identified for failure to conduct testing to determine torus-to-drywell vacuum relief check valve’s opening force under actual in-service conditions. Specifically, these four vacuum relief check valves were being cycled open and closed prior to measurement of their opening force. Operation of the valves immediately prior to testing has the potential to alter the amount of force required for subsequent operation, by releasing adhesion between the disc shaft packing and the shaft that could develop during the previous period of inactivity.

Description. On August 6, during the performance of surveillance N1-ST-Q5, “Primary Containment Isolation Valve Operability Test,” RB-to-torus vacuum relief check valve 68-05 closed position indicator did not extinguish when the valve was fully open. The valve was declared inoperable and was subsequently locked closed in accordance with TS 3.3.6. The closed position indicator limit switch was replaced and the valve stroke tested on August 8. The control room log stated that the valve initially hesitated and then stroked open very quickly. Following additional testing and engineering evaluation, the valve was declared operable.

As part of their evaluation of this operability determination, the inspectors reviewed surveillance test procedure N1-ST-SA6, “Drywell/Torus and Torus/RB Vacuum Relief Tests.” One purpose of this surveillance is to periodically measure the opening force required to operate the valves, as required by the Inservice Test Program and ASME Code. No issues with the August 6 testing of the RB-to-torus vacuum relief valves were identified. However, the inspectors noted a problem with the test methodology for the four torus-to-drywell vacuum relief valves. Immediately prior to measuring the opening force for these valves, the valves were each being cycled open and closed. This had the potential of altering the amount of force required to open the valve when the force measurement was being made (referred to as preconditioning), by releasing adhesion between the disc shaft packing and the shaft that could develop during the previous period of inactivity. However, the licensee considered the initial valve cycling to be acceptable preconditioning, since it was necessary to eliminate any differential pressure (d/p) that might exist between the torus and the drywell. Such a d/p would also alter the amount of force that would be required to open the valve. While the inspectors acknowledged this as a valid consideration, they determined that it would only be necessary to cycle the first valve to be tested, since any d/p that had existed would already have been eliminated before the remaining three were tested. By altering which valve was tested (and therefore preconditioned) first during the surveillance, as-found opening force measurements could be obtained on a rotating basis for all four valves.

The licensee entered this concern in the corrective action program as Deviation/Event Report (DER) NM-2005-3149. From this, the licensee concluded that cycling of torus-to-drywell vacuum relief valves subsequent to the first valve having been tested was unnecessary, and will be modifying the surveillance procedure accordingly.

Analysis. The performance deficiency associated with this issue was an inadequate surveillance procedure, in that the licensee failed to recognize that cycling the remaining three torus-to-drywell vacuum relief valves, after the first had already eliminated any d/p that may have existed, was unnecessary and constituted unacceptable preconditioning. The finding was greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of the procedure quality of a risk important surveillance and affects the cornerstone objective of providing reasonable assurance that physical design barriers (specifically, the primary containment) protect the public from radionuclide releases caused by accidents or events. 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 did not represent a degradation of the radiological barrier function provided for the control room, spent fuel pool, or standby gas treatment system, did not represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere, and did not represent an actual open pathway in the physical integrity of reactor containment or involve an actual reduction in defense-in-depth for the atmospheric pressure control or hydrogen control functions of the reactor containment.

Enforcement. 10 CFR 50, Appendix B, Criterion XI, “Test Control,” states, in part, “A test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures . . ." Contrary to the above, surveillance test procedure N1-ST-SA6, "Drywell/Torus and Torus/RB Vacuum Reliefs Test," Revision 00, dated July 2, 2004, did not demonstrate that the torus-to-drywell vacuum relief valves would perform satisfactorily in service, in that the valve opening force measurement was not made with the valves in their undisturbed, in service condition, but rather, immediately after the valves had been cycled open and shut. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (DER NM-2005-3149), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000220/2005004-02, Unacceptable Preconditioning of Torus-to-Drywell Vacuum Relief Valves During Surveillance Testing.

1R17 Permanent Plant Modifications (71111.17B - 10 Samples)

a. Inspection Scope

The inspectors reviewed ten permanent plant modifications to verify that the design bases, licensing bases, and performance capability of risk significant systems, structures, and components had not been degraded through plant change processes.

Plant changes were selected for review based on risk insights for the plant and included SSCs associated with the initiating events, barrier integrity and mitigating systems cornerstones. The inspection included interviews with plant staff, and the review of applicable documents including procedures, calculations, modification packages, engineering evaluations, drawings, corrective action documents, and the FSAR/USAR and TSs.

The inspectors verified that selected attributes were consistent with the design and licensing bases. These attributes included component safety classification, energy requirements supplied by supporting systems, instrument setpoints, and supporting electrical and mechanical calculations and analyses. Design assumptions were reviewed to verify that they were technically appropriate and consistent with the UFSAR. For selected permanent plant changes, the 50.59 screens or evaluations were reviewed as described in section 1R02 of the report. The inspectors verified that procedures, calculations and the UFSAR were properly updated with revised design information and operating guidance. The inspectors also verified that post-modification testing was adequate to ensure the SSC would function in accordance with its design assumptions. A listing of documents reviewed is provided in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 5 Samples)

a. Inspection Scope

The inspectors reviewed post-maintenance testing (PMT) procedures and associated testing activities for five 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:

- WO 04-08878-00, Feedwater/High Pressure Coolant Injection power supply modification to eliminate single point vulnerability (Unit 1).
- N1-ST-Q8B, Liquid Poison Pump 12 and Check Valve Operability Test (Unit 1).
- N1-ST-Q6C, Containment Spray System Loop 112 Quarterly Operability Test (Unit 1).
- N1-ST-M4B, Emergency Diesel Generator (EDG) 103 and PB 103 Operability Test (Unit 1).
- N1-ST-M6, CS Keep Fill System Verification Test (Unit 1).

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 6 Samples)

a. Inspection Scope

The inspectors witnessed performance of six 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-ST-M4A, EDG 102 and PB 102 Operability Test (Unit 1).
- N1-ST-Q1B, CS 121 Pump Valve and Shutdown Cooling System (SDC) Water Seal Check Valve Operability Test (Unit 1).
- N1-ST-Q1A, CS 111 Pump Valve and SDC Water Seal Check Valve Operability Test, with N1-ST-M6, CS Keep Fill System Verification Test (Unit 1).
- N1-ST-Q4, Reactor Coolant System Isolation Valves Operability Test (Unit 1).

- N2-OSP-CSH-Q@002, High Pressure CS Pump and Valve Operability and System Integrity Test (Unit 2).
- N2-ESP-ENS-Q731, Quarterly Channel Functional Test of LPCS/LPCI Pumps A, B, and C Auto Start Time Delay Relays (Unit 2).

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope (71114.06 - 1 Sample)

On August 8, the licensee conducted an Emergency Preparedness (EP) drill. The inspectors reviewed the drill scenario, applicable emergency plan implementing procedures (EPIPs), and emergency action levels (EALs.) The inspectors observed licensee performance during the drill including event classification, offsite authority notification, and dose assessment activities. Mitigation strategies and communications were observed. The inspectors noted that EP equipment and facilities were satisfactorily maintained in the emergency operations facilities.

The inspectors observed the post-exercise critique and also determined that the drill was appropriate in scope to be included in the EP performance indicator (PI) statistics. The site drill report and associated DERs which were generated were reviewed. Overall drill performance was reviewed against criteria contained in the Site Emergency Plan.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope (71122.01 - 10 Samples)

The inspector reviewed the most current Radiological Effluent Release Report to verify that the program was implemented as described in Radiological Effluent Technical Specification/Offsite Dose Calculation Manual (RETS/ODCM); reviewed the report for significant changes to the ODCM and to radioactive waste system design and operation; determined whether the changes to the ODCM were made in accordance with Regulatory Guide 1.109 and NUREG-0133 and were technically justified and documented; determined whether the modifications made to radioactive waste system design and operation changed the dose consequence to the public; verified that technical and/or 10 CFR 50.59 reviews were performed when required; and, determined whether radioactive liquid and gaseous effluent radiation monitor setpoint calculation methodology changed since completion of the modifications. The inspector determined that anomalous results reported in the current Radiological Effluent Release Report were adequately resolved. The inspector reviewed RETS/ODCM to identify the effluent radiation monitoring systems and its flow measurement devices; reviewed effluent radiological occurrence PI incidents for onsite follow-up; reviewed licensee self assessments, audits, and licensee event reports that involved unanticipated offsite releases of radioactive material; and, reviewed the FSAR/USAR description of all radioactive waste systems.

- The inspector walked down the major components of the gaseous and liquid release systems (e.g., radiation and flow monitors, demineralizers and filters, tanks, and vessels) to observe current system configuration with respect to the description in the FSAR/USAR, ongoing activities, and equipment material condition.
- The inspector reviewed several radioactive liquid and gaseous waste release permits, including the projected doses to members of the public.
- The inspector reviewed the records of any abnormal releases or releases made with inoperable effluent radiation monitors and reviewed the licensee's actions for these releases to ensure an adequate defense-in-depth was maintained against an unmonitored, unanticipated release of radioactive material to the environment.
- The inspector reviewed changes made by the licensee to the ODCM as well as to the liquid or gaseous radioactive waste system design, procedures, or operation since the last inspection. For each system modification and each ODCM revision that impacted effluent monitoring or release controls, the inspector reviewed the licensee's technical justification and determine whether the changes affect the licensee's ability to maintain effluents as low as is reasonably achievable (ALARA) and whether changes made to monitoring instrumentation resulted in a non-representative monitoring of effluents.
- The inspector reviewed a selection of monthly, quarterly, and annual dose calculations to ensure that the licensee had properly calculated the offsite dose from radiological effluent releases and to determine if any annual TS/ODCM (i.e., Appendix I to 10 CFR Part 50 values) were exceeded and, if appropriate, issued a PI report if any quarterly values were exceeded.
- The inspector reviewed air cleaning system surveillance test results and licensee specific methodology to ensure that the system is operating within the licensee's acceptance criteria. The inspector also reviewed surveillance test results and methodology the licensee uses to determine the stack and vent flow rates and verified that the flow rates are consistent with RETS/ODCM or FSAR values.
- The inspector reviewed records of instrument calibrations performed since the last inspection for each point of discharge effluent radiation monitor and flow measurement device and reviewed any completed system modifications and the current effluent radiation monitor alarm setpoint value for agreement with

RETS/ODCM requirements. The inspector also reviewed calibration records of radiation measurement (i.e., counting room) instrumentation associated with effluent monitoring and release activities and reviewed quality control records for the radiation measurement instruments.

- The inspector reviewed the results of the interlaboratory comparison program to verify the quality of radioactive effluent sample analyses performed by the licensee; reviewed the licensee's quality control evaluation of the interlaboratory comparison test and associated corrective actions for any deficiencies identified: and reviewed the results from the licensee's quality assurance audits and determined that the licensee met the requirements of the RETS/ODCM.
- The inspector reviewed the licensee's Licensee Event Reports, Special Reports, audits, and self-assessments related to the RETS/ODCM program performed since the last inspection. The inspector determined that identified problems were entered into the corrective action program for resolution. The inspector also reviewed deviation/event reports (DER) affecting RETS/ODCM.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

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

The inspectors performed a daily screening of items entered into the licensee's corrective action program as required by Inspection Procedure 71152, "Identification and Resolution of Problems." The review facilitated the identification of potentially repetitive equipment failures or specific human performance issues for follow-up inspection. This was accomplished by reviewing each issue report, attending daily screening meetings, and accessing the licensee's computerized database.

.2 Permanent Plant Change and 50.59 Review Activities

a. Inspection Scope

The inspectors noted that the licensee had recently put into place engineering management orders for special compensatory actions for the issuance of design changes. Previously, DER NM-2005-3118 was initiated on August 11, 2005, in order to document a self identified concern with the recent high frequency of problems encountered with implementing design changes at Nine Mile Point. This issue was categorized with a significance category code of one which requires a root cause analysis to be performed. The inspectors reviewed the proposed actions of this condition report along with several other condition reports associated with design changes at Nine Mile Point. The reviews were performed to identify whether

Constellation had been effective in identifying and resolving problems associated with the plant modification process and activities.

The inspectors reviewed issues that had been entered into the corrective action program to determine if the licensee had been effective in identifying problems associated with the 10 CFR 50.59 safety evaluation process. A sample of these issues was selected for further review during which the inspectors assessed the adequacy of the corrective actions which had been implemented or proposed for the selected issues.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

Review of the Institute of Nuclear Power Operations (INPO) 2004 Evaluation

a. Inspection Scope

The inspector reviewed the final report of the INPO 2004 evaluation of Nine Mile Point dated May 26, 2005. The evaluation was conducted during the weeks of October 18 and 24, 2004. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspective of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 14, 2005, the inspectors presented the inspection results to Mr. Tim O'Connor, and other members of his staff who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

J. Gerber, ALARA Supervisor
R. Godley, Manager, Operations
B. Holston, Manager, Engineering Services
J. Hutton, Director of Licensing
A. Julka, CEG, Director, Q&PA
T. Kulczycky, Reliability Engineering
T. Morgren, GS, Design Engineering
T. O'Connor, Plant General Manager
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

05000220/2005004-01	NCV	Failure to Manage Risk Associated With Maintenance on Power Board 11 Breaker Resulted in a Reactor Scram.
05000220/2005004-02	NCV	Unacceptable Preconditioning of Torus-to-Drywell Vacuum Relief Valves During Surveillance Testing.

Closed

NONE

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

<u>Section 1RO2: Evaluation of Changes, Tests, or Experiments</u>

<u>10 CFR 50.59 Safety Evaluations</u>

1998-097	ECCS Pump Performance Reconstitution
2001-001	Procedure Changes to Modify the Movement of Water in the Reactor Head Cavity During Disassembly for Refueling or Reassembly
2001-068	Substitute Position Feedback and Demand Signals with Operator Action to facilitate manual operation of the RCS Flow Control Valves
2002-001	Temporary Plugging the Vent Port of Solenoid Operated Valve SOV-39-05E
2003-003	CS Elimination of Water Leak Rate Requirements for CRS Torus Suction Valves
2004-002	Use of Potassium Iodine as the Comp Measure for Addressing Excessive Control Room Unfiltered Inleakage

<u>10 CFR 50.59 Screens</u>

AR 45489	Preventive Maintenance Requirements to be Changed from 18 Months to 24
EE 00360	CKV-39-04 Internal Parts Redesign
N1-OP-45	PCE 70273 Emergency Diesel Generators
N1-OP-45	PCE 69877 Emergency Diesel Generators
N1-ST-V19	Emergency Cooling System Heat Removal Capability Test at High Power
N1-03-028	CRD Pumps 11 and 12 Feeder Breakers Instantaneous Trip Setpoint Revision
N1-04-108	Revised Torus Pool Heat Up Analysis
N1-56-100	Emergency Condenser Eagle Timer Replacement
N2-03-069	SLCS Pump Discharge Relief Setpoint Margin
N2-04-184	Raise Undervoltage Alarm Setpoint to a Value Above the Minimum Operability Value of 130 VDC
N2-04-213	Change stroke time for RCIC Valves MOV 126 and 143
N2-05-010	Eliminate Single Point Vulnerability for Main Steam Tunnel Cooling
N2-OP-33	PCE 67149 High Pressure Core Spray
N2-OP-35	RCIC Pipe Fill
N2-ARP-01	PCE 67622 Operator Action to Bypass RCIC room temperature isolations

<u>Miscellaneous</u>

EPIP-Epp-15, Revision 6, Radioprotective Drug (KI) Administration
H21C-095 - Control Room LOCA Dose vs. Inleakage Using AST Methodology, 15 SCFH Bypass Leakage per MSIV and GL 91-18
Regulatory Guide 1.187 - Guidance for Implementation of 10 CFR 50.59
Regulatory Guide 1.195 Appendix A - Assumptions for Evaluating the Radiological Consequences of LWR Loss Of Coolant Accidents

Section 1R17: Permanent Plant Modifications

Permanent Plant Modifications

CTN2-55-802	ECCS Pump Performance Reconstitution
N1-01-058	CKV-39-04 Internal Parts Redesign
N1-03-028	CRD Pumps 11 and 12 Feeder Breakers' Instantaneous Trip Setpoint
N1-05-052	Replace Cables For Emergency Condenser Solenoid Valves
N1-56-100	Replacement of EC Timer Relays
N2-03-069	SLCS Pump Discharge Relief Setpoint Margin
N2-04-184	Raise Undervoltage Alarm Setpoint to a Value Above the Minimum Operability Value of 130 VDC
N2-55-758	DDC 2M11313 Revision of SBO Bases Document/Clarifies SBO Study
N2-OP-35	Revision 6 to RCIC Procedure For Filling Pipe With Pump
S15-72-ESW	ESW Pump Run-Out Calculation changes

Calculations & Analyses

S-15-72-ESWPMP, Revision 1, ESW Pump Run-Out concern with Drag Valve
S15-72-F001, Revision 4, Emergency Service Water IST Acceptance Criteria and Pump Curves
U/1 DDC N1-01-088, Revision 0, Change Tap Changer Setting on Transformers 101N(S)
U/2 Calculation A10.1-G-050, Revision 0, RCIC B/U Function
U/2 Calculation A10.1-J-038, Revision 02, Station Blackout
U/2 Calculation ES-268, Revision 0, Station Blackout-RCIC Pump/Turbine Room
U/2 Calculation EC-42, Revision 8, Verification of Adequacy of Division I Battery 2BYS*BAT2A
U/2 Calculation EC-43, Revision 8, Verification of Adequacy of Division I Battery 2SYS*BAT2B
U/2 Calculation EC-145, Revision 2, Verification of Adequacy of Division I Battery 2BYS*BAT2C

DERs & Condition Reports

CR 2002-2909
CR 2002-4518
CR 2003-1406
CR 2005-1505
CR 2001-3242
CR-2005-649
CR 2005-2904
CR-2005-3672
CR 2005-3673
CR 2005-3685
DER-2004-0856
DER-2004-4570

Drawings

1.774-001-268 Revision 5, High Pressure Core Spray Emergency Diesel Generator Power Supply
C-34812-C Sh.1 Revision 7, Remote Reactor Shutdown System Miscellaneous Instruments

C-34812-C Sh.2 Revision 4, Remote Reactor Shutdown System Miscellaneous Instruments
C-18017-C Sh.1 Revision 53, Emergency Cooling System
12177-ER-053-SK Revision 1, Circuit Breaker Trip Device Settings 125 VDC 2BYS*SWG002A

Miscellaneous

Disposition 02A Power Board 16 Coordination Study for Calc. 600VACPB16PDCS
Lesson Plan 02-OPS-001-263-2-01 Revision 5, Plant DC Electrical Distribution (BYS/BWS)

Procedures

PWM-PRO-0309 Revision 0, Special Test and Modification Functional Test Procedures
N1-RCPM-GEN-155 Revision 1, Load Testing of AK and ITE Breaker Trip Devices
N2-RCPM-GEN-V070 Revision 2, Protective/Auxiliary Relays and Timers

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

Nine Mile Point Nuclear Station - Unit 1 Annual Radioactive Effluent Release Report (2004)
Nine Mile Point Nuclear Station - Unit 2 Annual Radioactive Effluent Release Report (2004)
Nine Mile Point Nuclear Station, Nine Mile Point Unit 1 Off Site Dose Calculation Manual, Rev 25, February 2004
Nine Mile Point Nuclear Station, Nine Mile Point Unit 2 Off Site Dose Calculation Manual, Rev 25, January 2004
Quality and Performance Assessment Audit CHE-05-01-N, June 22, 2005

Nine Mile Point Procedures:

N1-RSP-13Q, Revision 7, Stack Radiation Monitor Quarterly Calibration Check and Channel Check
N1-RSP-11A, Revision 7, Calibration of the SW Discharge Monitor
N1-RSP-12Q, Revision 6, Calibration of the RB Vent Radiation Monitor
N1-RSP-9C, Revision 6, Instrument Channel Calibration of Emergency Condenser Vent Radiation Monitors
N1-RSP-14A, Revision 4, Calibration of the Rad Waste Discharge Monitor
N1-RSP-6c, Revision 8, Control Room Ventilation Radiation Monitor Instrument Channel Calibration
N1-TTP-040, Revision 2, Circulating Water Pump Performance Test
N1-ISP-112-A0001, Revision 3, Stack Gas Monitor Calibration
N1-ISP-085-001, Revision 2, Radwaste Discharge to Tunnel Radiation Monitor Instrument Calibration
N1-ISP-085-002, Revision 1, Liquid Radwaste Effluent Line
N1-ISP-112-005, Revision 2, Stack Flow Instrument Calibration
N1-ISP-112-004, Revision 2, Off-Gas Radiation Monitor (NUMAC) Instrument Channel Calibration
N1-IMP-999-039, Revision 2, Process Monitor H.V. and Discriminator Setting
N1-ISP-112-010, Revision 3, Stack Gas Process Radiation Monitor Channel Calibration
N1-ISP-077-005, Revision 2, Off Gas Sample/System Flow Instrument Channel Calibration

N1-ISP-112-008, Revision 2, OGESMS Flow Instrument Calibration
N1-CSP-M341, Revision 5, Primary Containment Sampling and Analysis
N1-CSP-V342, Revision 3, Containment Purge Evaluation
N1-CSP-M350, Revision 4, Noble Gas Dose Calculations
N1-CSP-M351, Revision 2, Particulate, Iodine, and Tritium Calculations
N1-TSP-202-001, Revision 0, Testing of Unit 1 RB Emergency Ventilation System
N2-RSP-RMS-R116, Revision 6, Channel Calibration Test of the Liquid Radwaste Effluent Line Liquid Process Radiation Monitor
N2-RSP-RMS-R113, Revision 7, Channel Calibration Test of the SW Effluent Line Process Radiation Monitors 2SWP*CAB146A and 2SWP*CAB146B
N2-RSP-RMS-R112, Revision 5, Channel Calibration Test of the Cooling Tower Blowdown Line Liquid Process Radiation Monitor
N2-RSP-RMS-R103, Revision 5, Channel Calibration Test of the Standby Gas Treatment System Exhaust Process Radiation Monitor
N2-RSP-RMS-R109, Revision 4, Channel Calibration Test of the Main Control Room Ventilation Process Radiation Monitors
N2-ISP-OFG-R101, Revision 2, Refueling Cycle Channel Calibration of the Off Gas Header Flow to Stack Instrumentation
N2-ISP-RMS-R102, Revision 4, Operating Cycle Channel Calibration of the Gaseous Effluent Monitoring System
N2-ISP-LWS-Q001, Revision 5, Liquid Radwaste Discharge Flow to Lake Instrument Channel Functional Test
N2-ISP-CWS-Q001, Revision 3, Quarterly Functional Test of the Circulating Water Cooling Tower Blowdown Line Flow Instrument Channel
N2-ISP-SWP-Q012, Revision 4, Quarterly Functional Test of the SW Effluent Lines A and B Flow Instrument Channels
N2-ISP-SWP-R112, Revision 8, SW Effluent Lines A and B Flow Instrument Channel Calibration
N2-ISP-CWS-A101, Revision 4, Calibration Test of the Circulating Water Cooling Tower Blowdown Line Flow Instrument Channel
N2-RSP-RMS-P001, Revision 4, Source Check of the Liquid Radwaste Effluent Radiation Monitor
N2-CSP-CMS-@341, Revision 2, Containment Purge Evaluation

<u>NCS Corporation Report:</u> In-Place Testing of Nuclear Air Cleaning Systems 1HVC*FLT2A
In-Place Testing of Nuclear Air Cleaning Systems 1HVC*FLT2B

<u>Deviation Event Reports</u>

2003-2562	2003-3536	2003-4062
2003-2772	2003-3623	2004-3896
2003-2891	2003-4026	2005-0451
2003-2923	2003-4027	2004-2934
2003-2944	2003-5168	2004-3168
2003-2961	2003-2829	2004-2422
2003-2983	2003-2931	2004-2801
2003-3294	2003-3097	2004-2299
2003-3427	2003-3722	2004-2309
2003-3430	2003-3784	2004-2069

2004-1697
2003-4209
2003-4316
2003-4347
2003-4356
2003-4378
2003-4560
2003-4569
2003-4665
2003-4680
2004-0256
2004-0439
2004-0546
2004-0856
2004-0873
2004-1696
2004-1712
2004-2260
2004-2302
2004-2697
2004-3015
2004-3091
2005-0491
2005-1289
2004-0628
2004-3186
2004-3350
2004-3395
2004-3696
2004-3936
2004-3944
2004-3948
2004-4414
2004-4432
2004-4524
2004-4574
2004-4627
2004-4748
2004-4756
2004-4917
2004-5043
2004-5354
2004-5503
2004-5573
2004-5713
2005-0134
2005-0155
2005-0962
2005-1708
2005-1754
2005-1897
2005-2007
2005-2293
2005-2356
2005-2448
2004-5027
2004-5111
2004-0473
2004-0542
2005-2010
2005-2015
2005-2186
2005-2191
2005-2357
2005-2385
2005-2584
2005-2636
2005-2714
2005-2774
2005-0277
2005-0307
2005-2489
2004-0779
2004-0833
2004-0991
2004-1115
2004-4776
2004-4927

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LIST OF ACRONYMS

ADAMS	agencywide documents access and management system
ALARA	as low as is reasonably achievable
CFR	Code of Federal Regulations
DER	deviation/event report

d/p	differential pressure
EAL	emergency action level
EDG	emergency diesel generator
EP	emergency preparedness
EPIP	emergency plan implementing procedures
ERO	emergency response organization
FSAR	final safety analysis report
FW	feedwater
FWCI	feedwater coolant injection
HPCS	high pressure core spray
IMC	inspection manual chapter
INPO	Institute of Nuclear Power Operations
IR	inspection report
MR	maintenance rule
NCV	non-cited violation
NMPNS	Nine Mile Point Nuclear Station
NRC	U.S. Nuclear Regulatory Commission
ODCM	offsite dose calculation manual
PARS	publically available records
PI	performance indicator
PMT	post-maintenance testing
RB	reactor building
RBCLC	reactor building closed loop cooling
RETS	radiological effluent technical specifications
RHR	residual heat removal
SDC	shutdown cooling system
SDP	significance determination process
SRA	senior reactor analyst
SSCs	structures, systems, and components
SW	service water
TB	turbine building
TS	technical specification
UFSAR	updated final safety analysis report
URI	unresolved item