March 23, 2001

EA 01-074

Mr. R. G. Lizotte, Master Process Owner - Assessment
C/O Mr. D. A. Smith, Process Owner - Regulatory Affairs
Northeast Nuclear Energy Company (NNECO)
P.O. Box 128
Waterford, Connecticut 06385

SUBJECT: MILLSTONE UNITS 2 AND 3 - NRC INSPECTION REPORT 05000336/2000-014 AND 05000423/2000-014

Dear Mr. Lizotte:

On February 10, 2001, the NRC completed inspections at your Millstone Units 2 & 3 reactor facilities. The enclosed reports document the inspection findings which were discussed on March 5, 2001 with Messrs. E. Grecheck and R. Necci and other members of your staff.

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

Based on the results of these inspections, the inspectors identified one Unit 2 issue and one Unit 3 issue, both of which were determined to be of very low safety significance (Green). The Unit 2 issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating the Unit 2 issue as a Non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of these inspection reports, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Millstone facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Robert J. Summers, Acting Chief
Projects Branch 6
Division of Reactor Projects

Docket Nos.: 05000336, 05000423
License Nos.: DPR-65, NPF-49

Enclosures:
(1) NRC Inspection Report 05000336/2000-014
(2) NRC Inspection Report 05000423/2000-014

Attachment: NRC's Revised Reactor Oversight Process

cc w/encl:
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ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	05000336
License No.:	DPR-65
Report No.:	05000336/2000-014
Licensee:	Northeast Nuclear Energy Company
Facility:	Millstone Nuclear Power Station, Unit 2
Location:	P. O. Box 128 Waterford, CT 06385
Dates:	December 31, 2000 - February 10, 2001
Inspectors:	P. C. Cataldo, Resident Inspector, Unit 2 S. R. Jones, Senior Resident Inspector, Unit 2 T. A. Moslak, Health Physicist, Division of Reactor Safety (DRS) G. C. Smith, Senior Physical Security Inspector, DRS
Approved by:	Robert J. Summers, Acting Chief Projects Branch 6 Division of Reactor Projects Region I

SUMMARY OF FINDINGS

IR 05000336/2000-014; on 12/31/00-02/10/01; Northeast Nuclear Energy Company, Millstone Nuclear Power Station; Unit 2. Maintenance Rule Implementation.

The inspection covered a six-week period conducted by resident staff and regional inspectors. This inspection identified one Green finding, which was a non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP) [see Enclosure 3 for a description of the NRC Revised Reactor Oversight Process]. The significance of findings for which the SDP does not apply is indicated by “No Color” or by the severity level of the applicable violation.

A. <u>Inspector Identified Findings</u>

Cornerstone: Initiating Events

- **Green.** Due to inadequate initial evaluation of feedwater control (FWC) system failures, the licensee failed to identify that the FWC system had exceeded its reliability performance criteria in August 2000. As a result, goal setting and monitoring were not performed as required by paragraphs (a)(1) and (a)(2) of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” The degraded reliability of the FWC system was of very low safety significance because although FWC system failures increase the frequency of initiating events, potential FWC system failures are unlikely to prevent the feedwater system from performing its accident mitigation function of providing adequate feedwater to the steam generators for decay heat removal. This violation of 10 CFR 50.65 was classified as a Non-Cited Violation. (Section 1R12.1)

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

There were no violations identified by the licensee during this inspection.

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Report Details

SUMMARY OF UNIT 2 STATUS

The plant operated at essentially 100 percent power throughout the inspection period with the exception of minor power reductions for routine turbine control valve testing, and condensate pump maintenance.

1. REACTOR SAFETY
(Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity)

1R04 Equipment Alignments

a. Inspection Scope

The inspector performed a partial system alignment check of the "A" train of the reactor building closed cooling water (RBCCW) system. The inspector verified the system was correctly aligned in accordance with Surveillance Procedure (SP) 2611C, "RBCCW System Alignment Checks, Facility 1," Rev. 025-04, and system piping and instrumentation diagram 25203-26022 by:

- confirming valves used for flow balancing were correctly positioned
- confirming normal RBCCW flow through in-service safety-related components
- confirming inlet and outlet component isolation valves were correctly aligned for standby safety-related heat exchangers

b. Findings

No findings of significance were identified during this inspection.

1R05 Fire Protection

a. Inspection Scope

The inspector reviewed the licensee's Fire Hazard Analysis, the Appendix R Compliance Report, and applicable licensing and design bases for Fire Area I-1, Zone A, in the intake structure. This area contains the three safety-related service water pumps and their associated discharge strainers. The inspector verified the following fire protection attributes for this area:

- the number and type of smoke detectors was consistent with the fire hazard analysis
- the smoke detector alarm function in the control room was operational
- the fire extinguishers in the area were operational
- the fire-fighting strategy was consistent with available fire suppression equipment and fire hazards
- the licensee's control of transient combustible materials was adequate
- the fire detection and suppression capability for the area was consistent with the facility's licensing and design bases

b. Findings

No findings of significance were identified during this inspection.

1R06 Flood Protection Measures

a. Inspection Scope

The inspector reviewed Section 9.7 of the Millstone Unit 2 Final Safety Analysis Report to identify measures provided for protection from internal flooding caused by circulating water system failures. The inspector toured the main condenser area to evaluate the material condition of the circulating water system and of design features provided for protection from circulating water system failures. The inspector reviewed procedure IC 2440, "Circulating Water Pump Trips Functional Test," Rev. 0, which verifies the functional capability of the main condenser pit level switches to trip the main circulating water pumps following a circulating water system failure, and verified that the test had been successfully performed within the specified periodicity.

b. Findings

No findings of significance were identified during this inspection.

1R12 Maintenance Rule Implementation

.1 Feedwater Control System Failures

a. Inspection Scope

The inspector reviewed the licensee's implementation of the maintenance rule for the feedwater control (FWC) system. The inspector reviewed the following condition reports (CRs) to verify that the identified issues were correctly classified with respect to maintenance preventable functional failures based on Engineering Department Instruction 30710, "Maintenance Rule Functional Failures":

M2-00-0148	M2-00-0236	M2-00-0375	M2-00-0376
M2-00-0504	M2-00-0699	M2-00-1889	M2-00-2026
M2-00-2073	M2-00-2104	M2-00-2376	M2-00-2381
M2-00-2385	M2-00-2504	M2-00-3100	M2-00-3427
M2-00-3437	M2-00-3440		

These CRs documented a number of occasions where the feedwater regulating valves (FRVs) (1) repositioned due to relay buffer card failures, (2) repositioned due to relay buffer card replacements coupled with human performance errors, and (3) experienced valve position transmitter failures and other component failures.

b. Findings

The inspector identified that FWC system failures occurring since January 2000, had not been appropriately evaluated as maintenance rule functional failures (MRFFs). As a result, the licensee neither demonstrated that system performance was being effectively controlled by preventive maintenance nor monitored the FWC system against licensee-established goals to assure acceptable performance in accordance with 10 CFR 50.65. The multiple failures of the FWC system were of very low safety significance (Green), because, although FWC system failures increase the frequency of initiating events, potential FWC system failures are unlikely to prevent the feedwater system from performing its accident mitigation function of providing adequate feedwater to the steam generators (SGs) for decay heat removal.

Of the 18 CRs the inspector reviewed, the licensee had identified the following two CRs as MRFFs:

CR No.	Event Date	Description	Plant Impact
M2-00-2385	8/29/2000	#2 FRV position appeared to fail high due to position indicator failure	Actual SG water level did not change
M2-00-3440	12/14/2000	Inadequate control of work to replace relay buffer cards caused plant transient	Full closure of the #2 FRV, and loss of two-thirds of the normal margin to the reactor trip setpoint for low SG water level

However, the inspector questioned the adequacy of the licensee's MRFF evaluations for eleven of the remaining CRs. The licensee reevaluated the issues relative to their MRFF criteria. Of the eleven CRs, the licensee reclassified the following four as MRFFs:

CR No.	Event Date	Description	Plant Impact
M2-00-0504	2/28/2000	#2 FRV position step change occurred from 67% to 73.5% due to position indicator failure;	Actual SG water level did not change
M2-00-2073	7/20/2000	#1 SG level indicator was reading high	Occasional plant process computer alarms
M2-00-2376	8/28/2000	Failed relay buffer card caused plant transient	#1 SG level indication increased, minor average thermal power reduction, and associated FRV partial closure; plant was later stabilized

CR No.	Event Date	Description	Plant Impact
M2-00-2504	9/11/2000	# 2 FRV controller pushbutton failed to shift FWC to manual	SG water level did not change; cause of pushbutton failure was never identified

Based on the sequence of the MRFFs described above, the four additional MRFFs substantiated by the licensee caused the FWC system to exceed the licensee's reliability performance criterion of less than 3 MRFFs per rolling 24 month period in August 2000. Due to inadequate initial evaluation of these CRs, the licensee failed to identify that they were required to monitor the FWC system performance against performance goals.

The inspector evaluated the FWC system failures under the NRC's Significance Determination Process (SDP). Of the six failures classified as MRFFs, the overall impact on plant operations was (1) control panel indication changes, (2) plant process computer alarms, (3) unanticipated reactor thermal power changes, (4) unexpected FRV position changes, including full closure, and (5) actual steam generator water level transients that prompted operator action to avoid a condition requiring a reactor trip. The inspector concluded that, although FWC system failures increase the frequency of initiating events, potential FWC system failures are unlikely to prevent the feedwater system from performing its accident mitigation function of providing adequate feedwater to the SGs for decay heat removal. Therefore, the SDP classified this condition as one of very low safety significance (Green).

10 CFR 50.65, paragraph (a)(1), requires, in part, that the performance or condition of structures, systems, and components (SSCs) be monitored against established goals to provide reasonable assurance that such SSCs are capable of performing their intended functions.

10 CFR 50.65, paragraph (a)(2), specifies that monitoring as specified in paragraph (a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance such that the SSC remains capable of performing its intended function.

Contrary to the above, the licensee failed to demonstrate that the performance of the FWC system had been effectively controlled through the performance of appropriate preventive maintenance. Specifically, the FWC system had experienced four failures of functions the licensee included within the scope of the maintenance rule between February and August 2000, which exceeded the performance criterion the licensee had established to demonstrate effective control of system performance through preventive maintenance. In addition, after the performance criterion had been exceeded, the licensee failed to monitor the performance of the FWC system against established goals, as required by paragraph (a)(1) when the provisions of paragraph (a)(2) were not met. Therefore, these failures are considered a violation of 10 CFR 50.65, paragraphs (a)(1) and (a)(2). This violation is more than minor because the degraded reliability of the FWC system had a credible impact on plant safety in that it potentially increased the frequency of initiating events. The violation is being treated as a Non-Cited Violation **(NCV 05000336/2000014-01)**, consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65 FR 25368). This violation is documented in the licensee's corrective action program as CR-01-01585. (EA 01-074)

.2 "A" Service Water (SW) Pump Discharge Strainer Differential Pressure Cell Failure

a. Inspection Scope

The inspector reviewed the licensee's maintenance rule implementation for the "A" SW pump discharge strainer differential pressure instrument failure documented in condition report CR-01-00811. The inspector verified that the condition was correctly classified with respect to maintenance preventable functional failures based on Engineering Department Instruction 30710, "Maintenance Rule Functional Failures." Since the service water system is currently in maintenance rule (a)(1) status, the inspector confirmed that the licensee had initiated appropriate actions to document and evaluate this latest functional failure against the (a)(1) action plan for the "A" train service water system.

b. Findings

No findings of significance were identified during this inspection.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

Control of Condenser Waterbox Outlet Isolation Valve Preventive Maintenance

a. Inspection Scope

The inspector reviewed work controls implemented under work order M2-99-06106, which involved preventive maintenance activities on 2-CW-11B, the "A" main condenser, "B" waterbox motor-operated outlet isolation valve. The review was conducted following the licensee's identification that scheduled work involved manipulating valve limit switches and that manipulation of the limit switches would have tripped a running circulating water pump and caused an unexpected plant transient. The inspector verified that the circulating water system is not considered a risk significant system relative to the maintenance rule, 10 CFR 50.65, and therefore would not have received an on-line risk review under MP-20-WM-FAP02.1, "Conduct of On-Line Maintenance." The inspector verified the adequacy of administrative controls and procedural requirements associated with maintenance activities as detailed in procedure MOV 1201, "Limitorque Operator Preventive Maintenance." Also, the inspector verified that the licensee initiated appropriate corrective actions once the potential circulating water pump trip was identified.

b. Findings

No findings of significance were identified during this inspection.

1R17 Permanent Plant Modifications

4160V Electrical Crosstie Project for Units 2 and 3

a. Inspection Scope

The inspector reviewed implementation, testing, and document update activities associated with Design Change Record M3-99039. This modification involved the installation of a new 4KV electrical connection from Unit 3 to Unit 2 in order to replace the current connection from Unit 1. The modification was designed to satisfy the following requirements:

- 10 CFR 50.63, "Loss of Alternating Current Power;"
- 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 17., "Electric Power Systems;" and
- 10 CFR Part 50, Appendix R, "Fire Protection Program."

The NRC previously completed design reviews, which were documented in NRC Inspection Report (IR) 05000336/2000-006 and 05000423/2000-006, dated May 5, 2000, and Inspection Report 05000423/2000-008, dated August 11, 2000. The inspector verified that concerns regarding design load profiles, transformer cable sizing, and breaker controls, described in IR 05000336/2000-006 and 05000423/2000-006, had been adequately addressed in subsequent revisions to the modification package.

The inspector completed the following activities:

- Verified that the licensee had adequately performed risk assessments to determine the relative impact of the testing on plant operations, and included appropriate technical specification (TS) revisions to support the testing.

- During pre-job briefs, verified that licensee personnel, including control room operators, were briefed regarding expected plant conditions throughout the testing, and expectations regarding the termination of testing due to unanticipated events.

- Observed the event-free performance of selected sections of SPROC-ENG00-S-01, "Millstone 4KV Cross Tie Test (IPTE)."

- Evaluated the adequacy of the licensee's safety evaluation performed in support of SPROC-ENG00-S-01, "Millstone 4KV Cross Tie Test (IPTE)."

- Verified that selected acceptance criteria were met through performance of the testing, and that these criteria were consistent with the applicable licensing and design bases.

- Verified during testing that the station blackout diesel could supply specific Unit 2 loads within one hour.

- Verified that the licensee had initiated applicable changes to licensing and design basis documents, such as the Final Safety Analysis Report.

- Verified that the licensee had revised affected procedures by reviewing the following sample of emergency, normal, and abnormal operating procedures:

  Unit 2

  - ARP 2590, “Alarm Response for Control Room Panel, C-08"
  - OP 2343, “4160 Volt Electrical System”
  - EOP 2528, “Loss of Offsite Power/Loss of Forced Circulation”
  - EOP 2530, “Station Blackout”

  Unit 3

  - OP 3346, “Station Blackout Diesel”
  - EOP 3501, “Loss of All AC Power (Mode 5, 6, and Zero)”
  - EOP 35 ECA 0.0, “Loss off All AC Power”

b. Findings

No findings of significance were identified during this inspection.

1R19 Post Maintenance Testing

“A” Service Water Pump Discharge Strainer Differential Pressure Switch Replacement

a. Inspection Scope

The inspector reviewed the post-maintenance testing associated with work orders M2-01-00794 and M2-01-00848, which involved the replacement of the “A” Service Water (SW) Pump discharge strainer differential pressure switch. The inspector verified that the post-maintenance tests for the applicable work orders were adequate, given the scope of the replacement activities, and provided adequate assurance that the SW pump discharge strainer met its design bases.

b. Findings

No findings of significance were identified during this inspection.

1R22 Surveillance Testing

.1 In-Service Testing

a. Inspection Scope

The inspector reviewed in-service testing (IST) results from a test conducted on January 9, 2001, and contained in OPS Form 2601A-7, “RWST Header Isolation Remote Position Indication IST.” The review included an assessment of the testing methods and a review of the acceptance criteria to ensure their consistency with the applicable licensing and design bases.

b. Findings

No findings of significance were identified during this inspection.

.2 Surveillance Testing

a. Inspection Scope

The inspector reviewed surveillance test results performed in accordance with the following procedures:

SP 2403EA	“RWST Channel A Level Calibration”
SP 2402M	“Functional Test of Steam Generator Level and Auto-Aux. Feedwater Initiation Logic”

The inspector verified that test results satisfied the applicable acceptance criteria, and that performance of the test adequately demonstrated equipment operability and capability to perform the intended safety function.

b. Findings

No findings of significance were identified during this inspection.

.3 Reactor Coolant System Sampling

a. Inspection Scope

The inspector reviewed reactor coolant system sampling performed in accordance with SP 2831, “Reactor Coolant Gross Specific Activity Determination.” The inspector verified that test results satisfied the acceptance criteria of Technical Specification 3.4.8, “Reactor Coolant System Specific Activity,” and that performance of the test adequately demonstrated required fuel cladding integrity.

b. Findings

No findings of significance were identified during this inspection.

1R23 Temporary Plant Modifications

Leak Sealant Repair of Valve 2-FW-41B

a. Inspection Scope

The inspector reviewed the licensee's leak sealant repair of the #2 steam generator feedwater regulating valve bypass valve, 2-FW-41B. The repair was initiated by the licensee to repair body to bonnet steam leaks that had developed during plant operation. The inspector reviewed the licensee's controls and implementation of the sealant repair under work order M2-00-19087, and the following procedures:

- C MP 718B, "Leak Sealing Procedure"
- C MP 715A, "General Practices for Threaded Fasteners"
- CEN 106B, "Determination of Torque Values for Pressure Retaining Fasteners"

The inspector verified that the leak repair activities maintained the operability and structural integrity of valve, 2-FW-41B, which is credited in achieving main feedwater isolation in the event of a main steam line break.

b. Findings

No findings of significance were identified during this inspection.

2. RADIATION SAFETY

Occupational Radiation Safety [OS]

2OS2 ALARA Planning and Controls

a. Inspection Scope

During the period January 8-12, 2001, the inspector conducted the following activities to evaluate the effectiveness of various controls to minimize and equalize personnel exposure for tasks conducted during power operations at Units 2 and 3, and those radiological controls that will be implemented during the Unit 3 refueling outage.

The inspector reviewed pertinent information regarding cumulative exposure history, current exposure trends, and ongoing activities in order to assess the licensee's effectiveness in establishing exposure goals in keeping actual personnel exposure as low as is reasonable achievable.

Performance of selected work groups was observed at both units. The inspector reviewed the associated exposure controls, attended pre-job briefings, and observed the de-sludging of the Unit 2 “A” Aerated Waste Tank and the replacement of the Unit 3 “B” Seal Injection Filter. For these tasks, the inspector interviewed selected workers on their knowledge of the relevant radiation work permit, electronic dosimetry set points, and job-site radiological conditions.

The inspector attended an outage ALARA Group meeting and reviewed the associated exposure controls specified in ALARA Reviews (AR) for selected jobs to be conducted during the Unit 3 refueling outage including:

Reactor Disassemble/Reassemble (AR3-01-01)
Steam Generator Eddy Current Inspection (AR3-01-02)
Steam Generator Secondary Side Sludge Lancing/Upper Bundle Flush (AR3-01-03)
In-Service Inspection (AR3-01-05)
Snubber Inspection and Repair (AR 3-01-06)
Mechanical Preventative & Corrective Maintenance Activities (AR3-01-09)
Valve Repair (AR3-01-11)
Outage Support - Scaffolding/insulation/Shielding (AR 3-01-13)
Replace Fuel Transfer Equipment (AR 3-01-15)

Interviews were conducted with cognizant engineering management regarding the status of temporary shielding to be installed on the Unit 3 steam generators to support outage activities, and shielding that is to be installed on the Unit 2 letdown line.

Independent radiation surveys were performed in the radiological controlled areas (RCA) of the Unit 2 and 3 auxiliary buildings, fuel handling buildings, and radwaste processing/storage areas to confirm posted survey results and assess the adequacy of radiation work permits and associated controls. Keys to technical specification "Locked High Radiation Areas" were inventoried and these areas were verified to be properly secured and posted during plant tours.

The effectiveness of various management controls were evaluated by reviewing the actions associated with a self-assessment (MP-SA-00-071) of the RCA access control program, various Nuclear Oversight Field Observations, and with recent Nuclear Oversight Verification Panel reports.

The inspector reviewed the following condition reports (CR) relating to the control of personnel exposure and work activities to determine if the issue was identified in a timely manner and that appropriate actions were taken to evaluate and resolve the issue.

M3-01-0013
M3-00-3120
M3-00-3476
M3-01-0069
M3-00-3484
M3-00-3504
M2-00-2844
M2-00-2781

b. Findings

No findings of significance were identified during this inspection.

3. SAFEGUARDS

Physical Protection [PP]

3PP4 Security Plan Changes

a. Inspection Scope (71130.04)

An in-office review was conducted of changes to the Physical Security Plan, identified as Revision 38, submitted to the NRC on June 12, 2000, in accordance with the provisions of 10 CFR 50.54(p). The inspector confirmed that the changes were made in accordance with 10 CFR 50.54(p), and did not decrease the effectiveness of the plan.

b. Findings

No findings of significance were identified during this inspection.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspector reviewed the licensee's first three quarters of performance indicator (PI) data submitted to the NRC for both Reactor Coolant System Activity and Reactor Coolant System Leakage. The inspector verified that the submitted data was consistent with plant records, which included: (1) a chemistry database where dose-equivalent iodine sample results are recorded, (2) plant process computer reports for identified reactor coolant system leakage, and (3) daily surveillance procedures where identified leak rates are recorded. The inspector verified that no significant problems or adverse conditions had been identified by the licensee relative to the coolant activity and leakage collection methods or compliance with applicable technical specification acceptance criteria.

b. Findings

No findings of significance were identified during this inspection.

4OA6 Meetings, including Exit

.1 Resident Inspector Exit Meeting

The inspectors presented the inspection results to the Vice President - Generation and the Vice President - Nuclear Technical services and other members of the licensee management at the conclusion of the inspection. The licensee acknowledged the findings presented.

ITEMS OPENED AND CLOSED

Opened

None

Opened and Closed During this Inspection

050000336/2000-014-01	NCV	Failure to Demonstrate Adequate Feedwater Control System Performance as Required by the Maintenance Rule (1R12.1)

Previous Items Closed

None

Previous Items Discussed

None

LIST OF ACRONYMS USED

AR	ALARA Reviews
CRs	condition reports
FRVs	feed regulating valves
FWC	feedwater control
IR	inspection report
IST	in-service testing
MRFFs	maintenance rule functional failures
PI	performance indicator
RBCCW	reactor building closed cooling water
RCA	radiological controlled areas
SDP	significant determination process
SGs	steam generator
SP	surveillance procedure
SSCs	structures, systems and components
SW	service water
TS	technical specification

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 05000423

License No.: NPF-49

Report No.: 05000423/2000-014

Licensee: Northeast Nuclear Energy Company

Facility: Millstone Nuclear Power Station, Unit 3

Location: P. O. Box 128
Waterford, CT 06385

Dates: December 31, 2000 - February 10, 2001

Inspectors: A. C. Cerne, Senior Resident Inspector, Unit 3
B. E. Sienel, Resident Inspector, Unit 3
T. A. Moslak, Health Physicist, Division of Reactor Safety (DRS)
G. C. Smith, Senior Physical Security Inspector, DRS

Approved by: Robert J. Summers, Acting Chief
Projects Branch 6
Division of Reactor Projects
Region I

SUMMARY OF FINDINGS

IR 05000423/2000-014; on 12/31/00-02/10/01; Northeast Nuclear Energy Company, Millstone Nuclear Power Station; Unit 3. Operability Evaluations

The inspection covered a six-week period, conducted by resident and regional inspectors. This inspection identified one Green finding. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP) [see Enclosure 3 for a description of the NRC Revised Reactor Oversight Process]. The significance of findings for which the SDP does not apply is indicated by "No Color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- **Green.** Operators failed to recognize that the "B" train of service water configuration rendered the train inoperable and therefore, were not tracking the inoperability against the 72 hour allowed outage time for Technical Specification limiting condition for operation 3.7.4, Service Water System, which is applicable in Modes 1 through 4. Operability Determination (OD) MP3-026-01 was written to justify operability of the "B" train of service water with the discharge check valve for one of the two pumps in the train missing a portion of its soft seat. One of the compensatory measures for this OD required that the "B" train of service water be operated with the "D" pump in the "lead" mode for a sequenced restart to prevent a drain-down of the service water system following a loss of power. However, on February 3, 2001, during Mode 4 (Hot Shutdown) conditions, the inspector identified that the above compensatory measure was not being followed. Instead, the "B" pump controls were in lead, as indicated on the main control board. Operators restored the compensatory measure upon identification of the problem. This condition apparently existed since the previous day, when the plant was at power. The time that the compensatory measures were not in affect did not exceed the TS allowed outage time one train of service water being inoperable. This condition was found to be of very low safety significance (Green) due to the fact that only one train of service water was affected for less than the TS allowed outage time for the plant conditions, and this condition would not have prevented the plant from being maintained in hot shutdown. (Section 1R15)

B. Licensee Identified Violations

There were no violations identified by the licensee during this inspection.

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Report Details

SUMMARY OF UNIT 3 STATUS

The plant began the inspection period on December 31, 2000, operating at approximately 95% power and coasted down to 77% power on February 2, 2001. On February 2 and 3, operators performed a controlled, manual shutdown of the reactor to begin the seventh planned refueling outage (3RO7). On February 8, the reactor was placed in Mode 6 (REFUELING), where it remained through the end of the report period on February 10, 2001.

1. REACTOR SAFETY
(Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity)

1R04 Equipment Alignments

a. Inspection Scope

During a period of time when modification activities were in progress in the engineered safety features (ESF) building cubicle housing the “A” train recirculation spray system (RSS) equipment, the inspector verified the correct alignment of the “B” train RSS components. The system walkdown included valve lineups, pump availability for an automatic start signal, and special piping flow path conditions (e.g., a loop seal configuration). The RSS system operating procedure, OP3306, as well as the “B” train valve lineup surveillance procedure, SP 3606.6, were used to confirm the operability of the system and correct flow path and component alignment.

The inspector noted the differences between the “B” and “D” RSS pump mini-flow piping/valve configurations and verified both lineups to be consistent with the applicable piping and instrumentation drawing, EM 112C-33. The inspector also reviewed condition reports addressing RSS “B” train equipment performance problems and discussed the technical specification requirements with the operations shift manager. The inspector discussed system performance and configuration with the RSS system engineer, including whether certain valve positions had changed as a result of previous design modifications and whether the affected component tagging and periodic surveillance activities reflected the implemented design changes.

b. Findings

No findings of significance were identified during this inspection.

1R05 Fire Protection

a. Inspection Scope

The inspector performed walkdowns of the south containment recirculation cooler cubicle (Fire Area ESF-1) in the ESF building and selected areas of the control building housing the control room (Fire Area CB-9) and above-floor area of the instrument rack room (Fire Area CB-11, Zone B). The inspector confirmed both the availability and condition of the fire detection and suppression equipment for these areas, as is described in the Millstone 3 Fire Protection Evaluation Report (FPER). The cognizant licensee fire protection engineer was interviewed with regard to specific fire area design features, exemptions, and the relevant FPER fire analysis.

Subsequently, the licensee initiated a condition report, CR-01-02224, to document errors identified in the FPER description of some of the inspected fire areas. These discrepancies did not adversely impact fire protection system operability in the affected areas. The inspector also noted that no compensatory measures were required for Fire Areas CB- 9 & 11, but that hourly fire roves were initiated when an ESF-1 door was blocked open. In accordance with the Unit 3 Technical Requirements Manual, such compensation is required for degraded or out of service equipment in these areas.

b. Findings

No findings of significance were identified during this inspection.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspector observed a simulator session that was conducted as part of licensed operator requalification training. The inspector observed operator use of emergency and abnormal operating procedures and alarm response procedures, in response to a loss of shutdown cooling caused by a loss of offsite power. The utilization of plant equipment operators for simulation of in-plant activities was noted during the observed session. The inspector discussed the scenario and training objectives with training personnel and attended the operating crew's self-critique following the scenario.

b. Findings

No findings of significance were identified during this inspection.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspector reviewed licensee actions taken in response to the following condition reports (CRs).

- M3-00-2466 Repetitive MRFF: False Radiation Monitor Trip Following Automatic Source Check
- M3-00-2845 Leak in Spent Fuel Pool Cooling (SFPC) System Piping
- M3-00-3221 Unisolable leak in SFPC System Drain Valve
- M3-00-3388 Charging Pump Cooling Isolated to the Running Charging Pump for 1.5 Minutes

For each CR identified, the inspector reviewed the applicable system's fourth quarter system health report, corrective actions taken in response to the equipment problem, maintenance rule functional failure (MRFF) determination, related CRs, and the a(1) action plan, where applicable. The inspector confirmed that the licensee appropriately tracked the occurrences against the systems' performance criteria.

Additionally, based upon the issuance of a condition report, CR-01-00230, in January 2001, regarding surveillance difficulties with a dual-function valve, 3RSS*MOV23B, the inspector verified the correct accountability of the valve inoperability against both the containment isolation function and the recirculation spray system availability. The inspector noted that an ineffective post-maintenance test plan for this valve was identified in the CR as an issue to be addressed by the licensee corrective action process, in order to preclude similar MRFF problems based upon surveillance test failures.

b. Findings

No findings of significance were identified during this inspection.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

a. Inspection Scope

The inspector noted a that the licensee had calculated a "yellow" online risk evaluation for work being performed on the "B" diesel generator, "B" diesel sequencer, and "B" service water train while the reactor was at 100 percent power. Since the unit typically is not in a yellow condition unless work is performed on the auxiliary feedwater system, this assessment was discussed with the responsible online risk reviewer to assess the cause of this condition and ensure this risk was understood by the licensee.

Also, the inspector reviewed the licensee assessment activities associated with changing conditions occurring in a "thru-wall" defect in an "A" train service water (SWP) system pipe line. After detecting the "thru-wall" defect, the licensee repaired it with a "non-code" clamp configuration. Further engineering review was required when subsequent ultrasonic testing revealed the flaw growth rate was greater than previously estimated (reference: CR M3-01-0051). The inspector confirmed that the licensee initiated dialogue with the NRC Office of Nuclear Reactor Regulation (NRR) for appropriate consideration of ASME Code Case N513 usage. The inspector examined the existing repair configuration, identified no SWP system leakage, and verified that operations personnel were conducting a visual inspection of this area each shift to verify no evident leakage.

Based upon a downward trend in the #1 seal leak-off flows for three reactor coolant pumps (RCP), as documented in CR M3-01-0088, the licensee initiated a "RCP Seal Leakoff Action Plan". The inspector noted that the observed problems and trend had first been detected as a high RCP leak-off flow, when the unit began its power coastdown (the boron concentration in the reactor coolant system reached levels where further dilution had negligible impact). The inspector verified that the licensee's plan was founded upon sequenced actions that were fully within allowed operational controls at power. The licensee consulted with the RCP supplier (Westinghouse) for both corrective measures and the limiting low flow criteria that would require an orderly plant shutdown to secure any affected RCPs. The inspector observed the licensee's implementation of the planned corrective actions, with the expected result that the affected RCP seal leak-off flow rates stabilized at levels above which they could be monitored without the need for a plant shutdown.

b. Findings

No findings of significance were identified during this inspection.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

During normal and backshift hours on February 2 and 3, the inspector observed operators perform portions of the planned shutdown and cooldown of the reactor to begin the seventh refueling outage. These activities included the removal of the plant from the grid, transfer to shutdown cooling using the residual heat removal system, and entry into Mode 5 (Cold Shutdown). The inspector discussed the evolutions with operators and observed the operators to confirm effective communication of plant conditions, appropriate understanding of, and response to, expected and unexpected alarmed annunciator conditions, and adherence to approved procedures and technical specification requirements. In the course of these plant evolutions, the following procedures were used.

- OP 3206 Plant Shutdown
- OP 3208 Plant Cooldown
- OP 3310A Residual Heat Removal System

b. Findings

No findings of significance were identified during this inspection.

1R15 Operability Evaluations

a. Inspection Scope

The following operability determinations (ODs) were reviewed. The inspector verified that: the engineering justifications for operability were sound; any required compensatory actions were implemented; and all applicable actions delineated in the Unit 3 technical specifications and technical requirements manual were met.

- MP3-008-00 Pin Hole Service Water (SWP) Leak Downstream of the Safety Injection Pump Cooler (Revision 1)
- MP3-026-01 Discharge Check Valve, 3SWP*V003, for “B” SWP Pump is Missing a Portion of Soft Seat Causing Backflow Through the Check Valve
- MP3-028-01 Based on Oil Sample Results From the “B” Containment Recirculation Spray (RSS) Pump, the Operability of the “A”, “C”, and “D” Pumps Has Been Questioned

- MP3-029-01 Latest Lube Oil Results Were Abnormal For Both Motor Bearings on RSS Pump, 3RSS*P1B

b. Findings

OD MP3-026-01 was written to justify operability of the “B” train of service water with the discharge check valve for one of the two pumps in the train missing a portion of its soft seat, causing backflow through the check valve. One of the compensatory measures for this OD required that the “B” train of service water be operated with the “D” pump in the “lead” mode for a sequenced restart to prevent a drain-down of the SWP system following a loss of power.

On February 3, 2001, during Mode 4 (Hot Shutdown) conditions, the inspector identified that the above compensatory measure was not in place. Instead, the “B” pump was in lead, as indicated on the main control board. Operators failed to recognize that this configuration rendered the “B” train of service water inoperable and were therefore not tracking the inoperability against the 72 hour limiting condition for operation for Technical Specification 3.7.4, Service Water System, which is applicable in Modes 1 through 4. Upon identification, with the plant still in Mode 4, the operators returned the “D” pump to lead, restoring the system to an operable status based upon the OD criteria, and placed caution tags on the pump switch to reinforce the compensatory measure. The licensee initiated condition report, CR-01-00926, to document the issue and provide a prompt evaluation.

At the end of the report period the licensee had not yet completed investigating the CR, but they believed that the incorrect swap of lead pumps occurred when the circulating pumps were taken out of service the previous day for the planned refueling outage. In order to determine a reasonable time during which the pumps were in this condition, the inspector reviewed operator logs and procedures. Inspector review of procedure, OP 3325A, Circulating Water, which was used to remove the circulating pumps from service, confirmed that the transfer of service water pumps in the lead position is directed by that procedure. The inspector also confirmed that the portion of the procedure used for back-flushing the circulating water bays, which had been performed earlier in the week, did not direct such transfers.

Therefore, using the time preliminarily identified in the CR, the inspector evaluated this condition for risk using phase 1 of the significance determination process (SDP) for both the reactor at power (when the condition was believed to be created), and for shutdown operations for the reactor in hot shutdown (when the condition was identified and corrected). Based on the condition duration identified, which was less than the TS allowed outage time of 72 hours for one train of service water being inoperable, this condition was found to be of very low safety significance (Green). This is due to the fact that only one train of service water was inoperable for less than the TS allowed outage time for both plant conditions, and the condition would not have prevented the plant from being maintained in hot shutdown, when one train of shutdown cooling (cooled by service water) was needed.

No other findings of significance were identified during this inspection.

1R17 Permanent Plant Modifications

.1 Groundwater Removal from the Engineered Safety Features (ESF) Building

a. Inspection Scope

The inspector reviewed the design change record (DCR) M3-00-004, including design change notice (DCN) DM3-00-0145, authorizing the replacement of the air motor-driven sump pumps with one deep-well electric submersible pump, and new collection sump and groundwater storage tank, for the removal of groundwater leakage into the recirculation spray system cubicles in the ESF building. Since portions of this modification involved non-safety related component interfaces with safety-related equipment, the inspector questioned the design aspects and quality assurance controls for such fabrication/construction activities. Periodically over the course of this inspection, the inspector examined the modification work in progress and verified adequate licensing amendment coordination with the NRC Office of NRR and appropriate Nuclear Oversight inspection of safety-related activities. This design change modification continued to be implemented through the end of this inspection period, with completion scheduled prior to the unit restart from refueling outage 3R07.

b. Findings

No findings of significance were identified during this inspection.

.2 4160V Electrical Crosstie Project for Units 2 and 3

Refer to Section 1R17 of NRC Inspection Report 05000336/2000-014 for the documentation of inspection activities relating to the 4160 Electrical Crosstie Project for Units 2 and 3.

1R19 Post-Maintenance Testing

a. Inspection Scope

During the plant power reductions manually controlled by operations personnel in preparation for taking the unit off-line for the 3R07 refueling outage, the plant was maintained at a plateau just below 50% power to conduct main steam safety valve (MSSV) testing. The inspector witnessed the conduct of testing activities on four of the seven MSSVs scheduled for surveillance prior to 3R07. The tests were performed using hydroset test devices and controlled as an infrequently performed test and evolution (IPTE), in accordance with the provisions of surveillance procedure, SP 3712G, Revision 007.

All seven MSSVs satisfied the “as found” criteria for safety valve lift set pressures, thereby requiring no additional valves to be tested. However, certain valves required adjustment of the “as left” set pressures, using hydraulic setting assist devices, in accordance with SP 3217G controls and criteria. The inspector observed and assessed the use of these controls, particularly with regard to one MSSV for which the proper “as left” setting could not be confirmed by testing. Discussions with the cognizant In-service test and maintenance personnel confirmed the licensee's plans to remove the MSSV in question during 3R07.

b. Findings

No findings of significance were identified during this inspection.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspector reviewed the following areas related to the refueling outage for conformance to technical specification requirements and approved procedures. Selected activities were verified for each evolution.

- Shutdown risk evaluations and implementation of recommendations
- Reactor shutdown and main turbine removal from the grid
- Reactor cooldown and placement of shutdown cooling in service
- Control of reactor vessel level while reduced for reactor head removal
- Outage configuration management with respect to outage risk
- Residual heat removal, spent fuel pool cooling, and containment closure controls
- Transfer of the new fuel assemblies, including implementation of special nuclear material accountability and double verification controls, from dry storage into Zone 1A of the new spent fuel pool storage racks
- Reactor engineering preparations (spent fuel pool and control room) for core offload activities, including fuel assembly transfer limits and controls

b. Findings

No findings of significance were identified during this inspection.

2. RADIATION SAFETY

Occupational Radiation Safety [OS]

2OS2 ALARA Planning and Controls

Refer to NRC Inspection Report 05000336/2000-014, Section 2OS2 for specific details.

3. SAFEGUARDS

Physical Protection [PP]

3PP4 Security Plan Changes

Refer to NRC Inspection Report 05000336/200-014, Section 3PP4 for specific details.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

.1 Safety System Functional Failures

a. Inspection Scope

The purpose of this inspection was to confirm that the information presented in the licensee's December 2000 Safety System Functional Failures performance indicator (PI) was complete and accurate. The inspector reviewed licensee event reports submitted between January 1 and December 31, 2000, to verify none should have been reported in the PI.

b. Findings

No findings of significance were identified during this inspection.

.2 High Pressure Injection System Unavailability

a. Inspection Scope

The purpose of this inspection was to confirm that the information presented in the licensee's December 2000 Safety System Unavailability PI for the high pressure safety injection (HPSI) system was complete and accurate. The inspector reviewed licensee event reports submitted during calendar year 2000; spot checked operational log entries for equipment out of service, as collated by the cognizant system engineers; and reviewed licensee Technical Evaluation M3-EV-00-0029, issued in July 2000, which addresses the unavailability monitoring requirements and PI calculation controls for the HPSI system. System engineering personnel were interviewed, as necessary, to discuss the unavailability data, particularly with regard to support system functionality and interconnecting system accountability.

b. Findings

No findings of significance were identified during this inspection.

4OA5 Other

.1 (Closed) LER 50-423/2000-03-00: Cable Routing for 3CHS*P3B Does Not Meet Fire Safe Shutdown Analysis.

Background:

This licensee event report (LER), dated October 26, 2000, identified that the “B” charging pump power cable was inadvertently located in the same fire area as the reactor plant component cooling water (CCP) pumps and heat exchangers. While this condition has existed since plant construction, the licensee’s fire analysis was revised several years ago to credit the subject charging pump, 3CHS*P3B, or the CCP system, which provides reactor coolant pump (RCP) thermal barrier cooling, also to provide RCP seal cooling following a fire. With vital support equipment located in the same fire area, this condition potentially compromised the plant design intended to prevent RCP seal damage that could lead to a small-break loss-of-coolant accident. The identified problem was reported as a condition outside the design basis of the unit.

At the time of the discovery of this condition, the licensee issued an operability determination supporting the position that the redundant, safety-related charging pump, 3CHS*P3A, was capable of being aligned to provide the RCP seal cooling function if the CCP pumps and “B” train charging pump both were adversely impacted by a fire. Other corrective measures implemented by the licensee involved enhancements to the fire prevention, detection, control, and suppression controls to the affected fire area, in order to minimize the possibility of the initiating event (i.e., major area fire) that could lead to the postulated RCP seal failure(s). Subsequently, the licensee relocated the power cable for pump, 3CHS*P3B, outside the subject fire area, bringing the plant into compliance with the design and licensing bases.

Risk Assessment:

There are several factors that influenced the risk significance of this condition. The “A” charging pump power cables are routed outside the fire area of concern, so this charging pump would be available to provide RCP seal cooling if the “B” charging pump was rendered inoperable by a fire. Another pump, the alternate “C” charging pump, also would not be affected by the postulated fire and would remain available. The “B” charging pump power cable was located very near a fire barrier water curtain and a considerable distance from any credible fire sources such as switchgear or the CCP pumps.

The inspector evaluated the design basis concern using Phase 2 of the NRC fire significance determination process (SDP) to establish the risk significance of this issue. An outline of the assumptions made follows.

- IF (initiating event frequency): From the Millstone IPEEE (Table 3.5-4), the fire frequency in this area is ~ 4.8E-3/year. IF ~ log(4.8E-3/yr) ~ -2.32

- FB ~ 0 (fire barrier): No credit was given for a fire barrier. This assumption is conservative due to the location of the cable and fire sources as discussed above.

- AS ~ 0 (automatic suppression): No credit was given because there is no AS in this area.

- MS ~ -1 (manual suppression): Normal operating state was assumed for the fire brigade.

Fire mitigation frequency (FMF): FMF=FB+MS+AS = -2.32+0+0+-1=-3.32.

From fire SDP, Table 5.6, the frequency is 1 per 1E-3 to 1E-4.

From fire SDP, Table 5.7, the “estimated likelihood rating” is Delta (D).

From fire SDP, Table 5.8, the remaining mitigating capability is 1 train (“A” charging pump and the availability (i.e., recovery with operator action) of the “C” charging pump; therefore, use column -3). Based on this, the condition was determined to be of very low safety significance (Green).

Conclusion:

The inspector evaluated the design basis concern documented in LER 50-423/2000-003-00 using the fire protection significance determination process. Based upon the defense in depth barriers for plant fire protection, only one of which was degraded by the identified problem, and also considering the availability of other multi-train equipment (“A” train charging pump and “C” swing charging pump) to mitigate the postulated, worst-case result of a major fire in the area, this event was found to be of very low safety significance (Green). This event did not constitute a violation of NRC requirements.

4OA6 Meetings, including Exit

.1 Resident Inspector Exit Meeting

The inspectors presented the inspection results to the Vice President - Generation and the Vice President - Nuclear Technical Services and other members of the licensee management at the conclusion of the inspection. The licensee acknowledged the findings presented.

ITEMS OPENED AND CLOSED

Opened

None

Opened and Closed During this Inspection

None

Previous Items Closed

50-423/2000-03-00	LER	Cable Routing for 3CHS*P3B Does Not Meet Fire Safe Shutdown Analysis (4OA5.1)

Previous Items Discussed

None

LIST OF ACRONYMS USED

AS	automatic suppression
CCP	reactor plant component cooling water
CRs	condition reports
DCN	design change notice
DCR	design change record
ESF	engineered safety features
FB	fire barrier
FPER	fire protection evaluation report
FMF	fire mitigation frequency
HPSI	high pressure safety injection
IF	initiating event frequency
IPTE	infrequently performed test and evolution
LER	licensee event report
MRFF	maintenance rule functional failure
MS	manual suppression
MSSV	main steam safety valve
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
ODs	operability determinations
PI	performance indicator
RCP	reactor coolant pump
RSS	recirculation spray system
SDP	significance determination process
SFPC	spent fuel pool cooling
SWP	service water
TS	technical specification

ATTACHMENT

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.