October 26, 2000

EA-00-247

Mr. Robert M. Bellamy
Site Vice President
Entergy Nuclear Generation Company
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, Massachusetts 02360-5599

SUBJECT: NRC's PILGRIM INSPECTION REPORT NO. 05000293/2000-008

Dear Mr. Bellamy:

On September 30, 2000, the NRC completed an inspection at your Pilgrim reactor facility. The enclosed report presents the results of that inspection. The results were discussed on October 10, 2000, with Mr. B. DiCroce, Mr. V. Oheim and other members of your staff.

This inspection was an examination of 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. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel. Based on the results of the inspection, there were no findings identified during this inspection.

The NRC identified one issue involving a discrepancy between the Emergency Operating Procedures and the Final Safety Analysis Report related to containment flooding in response to a design basis loss of coolant accident. The issue has been entered into your corrective action program and is discussed in the summary of findings and in the body of the attached inspection report. This issue was determined to involve a violation of NRC requirements. Due to the overall low risk significance of containment flooding, this violation was categorized at Severity Level IV. Consistent with the NRC Enforcement Policy, the violation is not cited. If you contest this non-cited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, 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 Pilgrim Nuclear Power Station.

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Sincerely,

/RA/

James Linville, Chief
Projects Branch 6
Division of Reactor Projects

Docket No.: 05000293
License No.: DPR-35

Enclosure: Inspection Report 05000293/2000-008

cc w/encl:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	05000293/2000-008
Licensee:	Entergy Nuclear Generation Company
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Inspection Period:	August 20, 2000, through September 30, 2000
Inspectors:	R. Laura, Senior Resident Inspector R. Arrighi, Resident Inspector B. McDermott, Senior Resident Inspector, Vermont Yankee R. Summers, Senior Project Engineer J. Furia, Senior Health Physicist B. Norris, Reactor Inspector
Approved By:	James Linville, Branch Chief Projects Branch 6 Division of Reactor Projects

<u>SUMMARY OF FINDINGS</u>

IR05000293-2000-008; on 08/20-09/30/2000; Entergy Nuclear Generation Company; Pilgrim Nuclear Power Station. Reactor Safety

The inspection was conducted by resident inspectors, a senior health physicist, a senior project engineer and a reactor inspector. This inspection identified one no color issue which was a noncited violation. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process.

- (NO COLOR) Pilgrim failed to perform a 10 CFR 50.59 safety evaluation to determine if a discrepancy between the EOPs and the FSAR involved an unreviewed safety question. The discrepancy related to the transition criteria for containment flooding in response to a design basis loss of coolant accident. Due to the overall low risk significance of containment flooding, this violation was categorized at Severity Level IV and was treated as a Non-Cited Violation. **(NCV 05000293/2000-08-01)**

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Report Details

SUMMARY OF PLANT STATUS

Pilgrim Nuclear Power Station began the period at 100 percent core thermal power. On August 21, 2000, the unit began an unplanned reactor shutdown due to indications of low bearing oil level in the "B" reactor recirculation pump motor. Inspection of the recirculation pump revealed a minor oil leak. After conferring with the pump vendor and verifying there was no damage to the pump, oil was added and the pump placed back in service. The unit was placed back on-line on August 25 and full power achieved on August 28, 2000. During the power ascension, the licensee performed a thermal backwash of the main condensers due to concerns with main condenser fouling. On September 29, 2000, the licensee discovered a feedwater/condensate flow mismatch and shut down the unit to investigate the cause. Investigation revealed damage to several of the "A" train fifth point feed water heater tubes.

1. REACTOR SAFETY

(Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity)

1R02 Evaluation of Changes, Tests, or Experiments (IP 71111.02)

a. Inspection Scope

The inspector reviewed the actions taken by the licensee in response to a 1998 unresolved item (URI 50-293/98-203-03) concerning the transition criteria in the Pilgrim Emergency Operating Procedures (EOPs) related to containment flooding as a mitigation strategy following a design basis accident loss of coolant accident (DBA-LOCA). The inspection included a review of the current EOPs, interviews with the engineering and training personnel familiar with the EOPs, the Boiling Water Reactor Owners' Group (BWROG) guidance documents, the safety evaluations, and Problem Reports (PRs). Specific documents reviewed include:

- EOP-01, Revision 5, Reactor Pressure Vessel (RPV) Control
- EOP-09, Revision 2, Primary Containment Flooding
- EOP-17, Revision 2, Alternate RPV Depressurization
- Safety Evaluation 87-170, Revision 0, Implementation of Revision 4AF EOPs
- Calculation PNPS-1-ERHS-XI.M-29, Revision 0, Radiological Consequences from Realistic Source Term LOCA with Containment Venting
- Pilgrim Final Safety Analysis Report (FSAR), Sections 6.5 (Safety Evaluation) and 14.5 (Postulated Design Basis Accidents)
- PR 98.9519 and PR 98.9541 - Related to the Implementation of Revision 4 EOPs
- NRC Safety Evaluation Report of the BWROG - Emergency Procedure Guidelines, Revision 4, dated September 12, 1988
- Summary for a Meeting with the NRC on the EOPs, dated September 18, 1991
- Summary of a BWROG Meeting on the EOPs, dated February 18, 2000

b. Findings

In 1998, the NRC noted that Pilgrim's EOPs would require the flooding of the primary containment building in the event of a DBA-LOCA. The flooding of the containment would be based on the inability to maintain the reactor pressure vessel (RPV) water level greater than the top-of-active-fuel (TAF). No consideration was given as to what emergency core cooling systems (ECCSs) were available for injection.

The Pilgrim EOPs were developed using Revision 4 of the BWROG emergency procedure guidelines (EPGs). The Pilgrim Updated Final Safety Analysis Report (UFSAR), Chapter 14.5.3, describes a DBA-LOCA as a break of one of the recirculation loop pipes inside of containment. Specifically, the UFSAR describes that the core spray system is used to remove the stored decay heat from the reactor core, and that the RPV is flooded to the height of the jet pump nozzles (2/3 core height). An inherent design feature of the BWR RPV is that, following a DBA-LOCA with minimal ECCS equipment available, the water level would be maintained at core height. This inherent feature allows for sufficient cooling of the fuel, but it implies that the water level would never be at the TAF during a DBA-LOCA.

The NRC Safety Evaluation for Revision 4 of the EPGs stated that "... each BWR licensee who wished to use Revision 4 of the EPG should assure that the EPGs will not impact its licensing basis." The EOPs directed operators to flood the primary containment, and vent the RPV to allow water to enter the vessel. The unfiltered venting of the RPV would have been performed irrespective of offsite dose rates. As such, containment flooding as mitigation for a DBA-LOCA constituted a change to the licensing basis and should have been evaluated in accordance with 10 CFR 50.59 to ensure that the change did not involve an unreviewed safety question (USQ).

The situation was created in the early 1990s, when Revision 4 of the BWROG EPGs was adopted into the Pilgrim EOPs. Since this issue was first identified, the BWROG revised the EPGs to include severe accident guidelines (SAGs). The current EPG/SAGs define a different criteria for initiation of containment flooding, including a definition for a minimum steam cooling reactor pressure vessel water level (MSCRWL). The MSCRWL is between the TAF and the 2/3 core height. If level is greater than the MSCRWL, and at least one injection subsystem is available and running, containment flooding is not required. Revision 2 to the EPG/SAGs (expected to be issued later this year) will change the containment flooding level to less than 2/3 core height; this will be consistent with the Pilgrim UFSAR. As of this inspection, Pilgrim initiated a change to the EOPs to change the transition point to 2/3 core height; the change is expected to be implemented by December 2000.

The inspector discussed this issue with individuals from the NRC's Office of Nuclear Reactor Regulation (NRR); specifically, whether containment flooding at the TAF constituted a USQ. The conclusion was that the changes to the EOPs to flood containment, although consistent with the Owners' Group recommendations, did constitute a USQ which required NRC approval. Specifically, when evaluated using the source term inside containment as assumed in the licensing basis analysis for a DBA-LOCA, the dose consequences with containment flooding would increase. Further, containment flooding created the possibility for new equipment malfunctions. Pilgrim provided a realistic source term dose assessment which revealed that the offsite dose as a result of containment flooding would be significantly less than the criteria contained in 10 CFR 100.

The above issue was determined to be more than minor and did not affect any cornerstone (i.e., no color) in accordance with NRC Manual Chapter 0610*, Appendix E. The failure to perform a safety evaluation for a change to the licensing basis and the failure to obtain NRC approval prior to implementation of a change involving a USQ constitutes a violation of 10 CFR 50.59. Due to the overall low risk significance of containment flooding, this violation of 10 CFR 50.59 was categorized at Severity Level IV and was treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). This finding was entered into the Pilgrim corrective action program as PR 00.2174.
(NCV 05000293/2000-08-01)

Based on the above, URI 50-293/98-203-03 is **closed**.

1R04 Equipment Alignment

a. Inspection Scope

1. The inspector conducted partial system walkdown of the reactor core isolation cooling (RCIC) and reactor water cleanup systems; this included verifying valve positions. The inspector verified that the RCIC steam line trip throttle valve was properly latched and sufficient lube oil was visible in the sump sight glass. Some evidence of minor oil leaks was evident but the licensee had already identified this leakage as indicated by a work control tag.

2. The inspector performed a complete system walkdown of the standby liquid control system (SLC). The walkdown involved reviews of the system's operating procedure (No. 2.2.24), quarterly surveillance procedure (No. 8.4.1), piping and instrument drawing (M249), and an in-plant verification of the system's alignment. The system lineup review included verifying that valves and electrical breakers were in the proper standby line-up condition and that the white lights and continuity meter in the control room indicated proper continuity of power for the squib valves. Instrumentation used to perform Technical Specification surveillance testing was discussed with the cognizant inservice test engineer. The inspector also reviewed open work orders and corrective action documents to assess any outstanding SLC equipment and/or component deficiencies.

b. Findings

There were no findings identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the implementation of the maintenance rule (10 CFR 50.65) as related to the following:

- Proper classification of equipment failures for the standby liquid control (SBLC) system during the previous 24 months. Various problem reports (PRs) were reviewed including PRs 98.9405 (SBLC accumulator T-223B found at 40 lbs) and PR 98.2285 (potential rework issue of relief valve PSV-1105A). The SBLC system is classified as an a(2) system.

- Proper classification of equipment failures for the primary containment system during the previous 24 months. Problem reports reviewed included: PR 99.9359 (MSIV AO-203-2C flow control valve internals missing) and PR 999358 (failed pressure drop test). The primary containment system is classified as an a(1) system.

- The residual heat removal (RHR) system was reviewed including the various operational modes such as low pressure coolant injection (LPCI), containment cooling and containment isolation. The inspector reviewed a listing of problem reports generated during the previous 1.5 years that involved equipment issues with the RHR system. Several potentially significant issues were reviewed including: PR 00.0934 (CK-1001-132 closure test), PR 00.9367 (torus leakage) and PR 99.9220 (MO-1001-29A and 29B body pressure relieving check valves not in IST program).

- Proper classification of equipment failures in the reactor core isolation cooling (RCIC) as documented in the corrective action process during the previous 1.5 years. The inspector reviewed PR 00.2462 (MO-1301-16 takes 3 to 4 jogs to get indication) and PR 99.9218 (valves 6-CK-62A and MO-1301-49 local leak rate test).

The inspector also reviewed the appropriateness of the associated a(1) or a(2) classification and the applicable a(1) corrective action plan.

b. Findings

There were no findings identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspector reviewed the following on-line maintenance work plans/activities to assess the adequacy of the licensee’s risk assessment process:

- Surveillance 8.M.2-2.5.3, HPCI Steam Line High Temperature

- Surveillance 8.M.2-2.1.10, A5/A6 under voltage testing
- Surveillance 8.M.2-2.6.1, RCIC Steam line High Flow Functional Test

The inspector reviewed the plans against the criteria contained in licensee procedures 1.5.21, "Integrated Scheduling Guidelines," and 1.5.22, Risk Assessment Process." The inspection also included a review of the risk assessments and contingencies established, and verification that the increase in risk was conveyed during the licensee's morning meeting and during operator shift turnover briefs. In addition, selected activities were observed to assure that the activities being conducted were in accordance with the controlling documents.

b. Findings

There were no findings identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events

a. Inspection Scope

The inspector monitored portions of the August 22, 2000, unplanned reactor shutdown due to a low bearing oil level in the "B" reactor recirculation pump. The licensee shut down the plant in accordance with procedure 2.1.5 section G, "Controlled Shutdown With One Recirculation Pump Out Of Service." The inspector monitored portions of the crew performance, observed a shift turnover during the plant shutdown, and reviewed the control room supervisor logs to assess operator response and verify that the shutdown was performed in accordance with approved procedures.

b. Findings

There were no findings identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspector reviewed and/or observed portions of the following post maintenance tests to ensure that test activities were adequate to verify operability and functional capability of the system/component following maintenance.

- Test of the spent fuel pool cooling system skimmer surge tank level control valve per MR 19602686
- Test of the high pressure coolant injection system flow controller per MR 10001814
- Test of the "A" train core spray system full flow test valve, MO-1400-4A, per S9810232
- Test of the "A" reactor recirculation system motor-generator set per MR 10001779

b. Findings

There were no findings identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspector reviewed surveillance test procedures and associated testing activities to assess whether (1) the test preconditioned the components tested, (2) the effect of the testing was adequately addressed in the control room, (3) the system requirements were correctly incorporated into the test procedures and the test acceptance criteria were consistent with the technical specifications and Updated Final Safety Analysis Report, (4) the test equipment range and accuracy was adequate with proper calibration, (5) the test was performed in the proper sequence, and (6) the test equipment was removed following testing and system configuration restored to a normal state of readiness.

The inspector reviewed and observed portions of the following surveillance tests:

- Procedure No. 8.M.2-2.5.3, HPCI Steam Line High Temperature
- Procedure No. 8.M.2-2.10.5, HPCI Auto-Isolation System Logic

The review also included an evaluation of the completed surveillance test data to verify the selected systems and components were capable of performing their intended safety functions and operational readiness.

The inspector reviewed the results of an "A" train emergency diesel generator (EDG) logic system functional test (LSFT). The inspector confirmed that all LSFT surveillance procedure acceptance criteria were met. Additionally, the inspector ensured that the LSFT test procedure was consistent with technical specifications and the Updated Final Safety Analysis Report. Lastly, the review also verified the "A" EDG was properly returned to an operational ready status.

b. Findings

There were no findings identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control (71121)

a. Inspection Scope

The inspector reviewed the access control program by examining the controls established for three exposure significant areas. Controls reviewed included: key control for locked high radiation areas; use of radiation work permits (RWPs) to control access to radiologically significant areas; survey frequency of posted areas; effects of changing plant conditions on dose rates; pre-job radiological briefings; postings; markings; dosimetry; and, surveys and alarm set points. These controls are used by the licensee to meet the requirements of 10 CFR 20.1601 and plant technical specifications. Areas selected were located throughout the radiologically controlled area (RCA) and included: (1) transverse in-core probe (TIP) room [very high radiation area]; (2) backwash receiver tank room [locked high radiation area]; and, (3) “B” flatbed filter room [high radiation area].

The inspector conducted job performance observations to evaluate radiation worker performance with respect to stated radiation protection work requirements. This also included verification of radiological controls, such as adequacy of surveys and radiation protection technician coverage. The inspector reviewed the RWPs utilized for these entries; attended the pre-job briefings presented to the workers by the radiation protection staff; observed controls present for access to these areas; and, reviewed alarm set points. On August 30, 2000, the inspector observed an entry to a very high radiation area (TIP room). The inspector attended the pre-job briefing, reviewed the job-specific RWP for the task (RWP No. 00-0006), alarm set point established for the work, and available surveys of the TIP room, verified appropriate access and key control to the area was maintained, and conducted direct field observation of the entry.

The inspector conducted tours of various locations within the radiologically controlled area of the plant, including the reactor, turbine, auxiliary and augmented off-gas buildings. The inspector verified the postings and access controls for two very high radiation areas, twelve locked high radiation areas, and five high radiation areas.

b. Findings

There were no findings identified.

2OS2 ALARA Planning and Controls (71121)

a. Inspection Scope

The inspector reviewed work performance during the current operating cycle. Areas reviewed included an evaluation of the use of engineering controls to achieve dose reductions; review of the use of low dose waiting areas; review of on-job supervision provided to workers; and, a review of individual exposures from selected work groups.

An evaluation of engineering controls utilized to achieve dose reductions, and analysis of licensee source term reduction plans was also conducted. This included a review of the current edition (dated May 31, 2000) of the Pilgrim Nuclear Power Station Dose Reduction Strategy.

The inspector conducted observations of radiation worker and radiation protection technician performance during high dose rate and/or high exposure jobs to determine if the training/skill level was sufficient with respect to the radiological hazards. Additionally, the inspector conducted reviews of work completed during a recent forced outage (began August 22, 2000) which involved work in the drywell, including under the reactor vessel.

b. Findings

There were no findings identified.

2OS3 Radiation Monitoring Instrumentation (71121)

a. Inspection Scope

The inspector reviewed the licensee's inventory of survey instruments and personnel contamination monitors and its program for daily source checking, maintenance and repair. This included walkdowns of the radiological instrumentation calibration and repair facility and instrument issue areas.

The inspector reviewed records of calibration for portable survey instruments utilized by radiation protection technicians to ascertain and document radiological conditions. Instruments utilized could measure alpha, beta, gamma and neutron radiation, as appropriate. The records reviewed were of all radiation protection department instrumentation calibrated during the month of August 2000. The inspector examined each type of instrument used via a review of calibration and source check records maintained by the licensee and reviewed documentation of calibration source traceability to the National Institute of Standards and Technology (NIST) and records of source usage. Additionally, the inspector observed repairs being made to a portal monitor (Aptec Model PMW-2) utilized at the main radiological control point.

b. Findings

There were no findings identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspector reviewed the first and second quarter 2000 data for the following NRC Performance Indicators (PIs) for Safety System Unavailability to verify that the licensee had characterized past events in accordance with the NRC endorsed criteria in NEI 99-02, "Regulator Assessment of Performance Indicator Guideline," Revision 0:

- Heat Removal System (RCIC)
- High Pressure Injection System (HPCI)
- Residual Heat Removal (RHR) System

b. Findings

There were no findings identified.

Concerning the Residual Heat Removal System PI, the inspector observed that Updated Final Safety Analysis Report (UFSAR) Section 10.4.2 states the primary method of decay heat removal will have sufficient cooling capacity such that the bulk temperature in the spent fuel pool will be at or below a nominal 125°F with the peak not exceeding 142°F at the point of maximum heat load occurring at the end of a spent fuel transfer or at any time during a full core off-load. However, the licensee has credited the fuel pool cooling system as "an alternate, NRC approved means of removing core decay heat" (see NEI 99-02, Revision 0, page 32, line 16) when it can maintain the bulk water temperature below 212 °F. This approach reduces the unavailability hours that otherwise accumulate when the licensee removes a train of RHR shutdown cooling from service during a refueling outage. Operation of the fuel pool cooling system, as credited for the PI, is not described in the UFSAR.

During this inspection, the licensee was not able to find an NRC approved license change, or other plant change approved under 10 CFR 50.59 requirements, that would support this use of the fuel pool cooling system. Therefore, the inspector could not conclude the licensee's accounting for unavailability hours in support of the Residual Heat Removal System PI was correct since the guidance stipulates an alternate, NRC approved means of removing core decay heat. The licensee entered this issue into the corrective action system as Problem Report 00.2834 to assess if this PI was being calculated correctly.

4OA6 Management Meetings

a. Exit Meeting Summary

The inspectors presented the inspection results to Mr.B. DiCroce, General Manager Plant Operation, and Mr. V. Oheim, Director Design and Engineering, and other members of licensee management at the conclusion of the inspection on October 10, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered propriety. No propriety information was identified.

ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

URI 50-253/98-203-03	Reconciliation of EOP Actions and UFSAR Assumption

Opened and Closed

NCV 05000293/2000-08-01	Failure to Perform a 10 CFR 50.59 Safety Evaluation

LIST OF ACRONYMS USED

ALARA	As Low As is Reasonably Achievable
CFR	Code of Federal Regulations
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EE	Engineering Evaluation
ENS	Emergency Notification System
EOP	Emergency Operating Procedure
HPCI	High Pressure Coolant Injection
LER	License Evaluation Report
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MSIV	Main Steam Isolation Valve
MR	Maintenance Request
NPSH	Net Positive Suction Head
NIST	National Institute of Standards and Technology
PMT	Post Maintenance Test
PR	Problem Report
RBCCW	Reactor Building Closed Cooling Water
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
SER	Safety Evaluation Report
SRO	Senior Reactor Operator
UFSAR	Updated Final Safety Analysis Report

ATTACHMENT 1
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.