January 29, 2001

Mr. Robert J. Barrett
Vice President, Operations-IP3
Entergy Nuclear Northeast
Indian Point 3 Nuclear Power Plant
Post Office Box 308
Buchanan, NY 10511

SUBJECT: NRC'S INDIAN POINT 3 INSPECTION REPORT NO. 05000286/2000-008

Dear Mr. Barrett:

On December 30, 2000, the NRC completed an inspection at the Indian Point 3 nuclear power plant. The enclosed report presents the results of that inspection. The results were discussed on January 11, 2001, with you and other members of your staff.

The 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.

No findings of significance were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room **or** from the Publicly Available Records (PARS) component of the 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 No.05000286
License No. DPR-64

Enclosure: Inspection Report No. 05000286/2000-008

cc w/encl:
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M. Kansler, Chief Operating Officer
J. Knubel, Vice President Operations Support
F. Dacimo, General Manager, Plant Operations
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	05000286
License No.	DPR-64
Report No.	05000286/2000-008
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point 3 Nuclear Power Plant
Location:	P.O. Box 308 Buchanan, New York 10511
Dates:	November 19 - December 30, 2000
Inspectors:	Peter Drysdale, Senior Resident Inspector Lois James, Resident Inspector
Approved by:	Robert Summers, Acting Chief Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000286/2000-008; on 11/19 -12/30/00; Entergy Nuclear Northeast; Indian Point 3 Nuclear Power Plant.

The inspection was conducted by the resident inspectors. The significance of most/all findings is indicated by the color (green, white, yellow, or red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). 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 description of the NRC Reactor Oversight Process is enclosed as Attachment 1 of this report.

A. There were no findings of significance identified during this inspection.

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

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Report Details

SUMMARY OF PLANT STATUS

The Indian Point 3 plant remained at full power from November 19 through December 17, 2000. On December 18, 2000, the plant was taken offline to conduct repairs on Main Generator Hydrogen Coolers. Plant power was raised to 100% on December 20, 2000, and has remained at full power since December 20, 2000.

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

1R04 Equipment Alignment

a. Inspection Scope (71111.04)

On December 12, 2000, the inspector performed a partial walkdown of the 32 component cooling water (CCW) System using Checkoff List COL-CC-1 "Component Cooling System," System Operating Procedure SOP-CC-1B "Component Cooling System Operation," and system flow diagrams 9321-F-27203 and -27513. During this inspection, the 31 CCW pump was out of service for preventive maintenance on the pump motor. The inspector verified the valve lineup in the common CCW pump discharge header.

b. Findings

There were no significant findings identified during this inspection.

1R05 Fire Protection

a. Inspection Scope (71111.05Q)

Quarterly Fire Protection Walkdown

The inspector conducted a fire protection tour of the Auxiliary Feedwater Pump room on November 24, 2000, to observe: (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational lineup, and operational effectiveness of fire protection systems, equipment and features; and (3) the material condition and operational status of fire barriers used to prevent fire damage or fire propagation.

Annual Observation of Fire Drill

On November 27, 2000, the inspectors performed the annual observation of a fire brigade drill to evaluate the readiness of the licensee's personnel to suppress fires. The drill fire was in the site engineering building located between the warehouse and the central access point. The licensee chose this location due to the potential to hinder access to both the central access control point and the warehouse, storage site of Appendix R equipment. The drill included the offsite fire department and full discharge of the water through the fire hoses. The inspectors observed various aspects of the fire brigade drill and assessed the effectiveness of the following attributes:

- Protective clothing/turnout gear was donned properly.
- Self-contained breather apparatus (SCBA) equipment were worn appropriately.
- Fire hose lines were capable of reaching the fire location.
- Fire hose lines were laid out without flow constrictions.
- Fire hoses were charged with water.
- The fire brigade entered the fire scene in a controlled manner.
- Sufficient fire fighting equipment was brought to the scene by the fire brigade.
- The fire brigade leader's directions were thorough, clear, and effective.
- Radio communications with the plant operators and between fire brigade members were efficient and effective.
- The pre-planned drill scenario was followed.
- The drill acceptance criteria were met.

b. Findings

There were no significant findings identified during this inspection.

1R11 Licensed Operator Requalification

a. Inspection Scope (71111.11)

On December 14, 2000, the inspector observed a portion of periodic operator requalification simulator training for licensed operators of Crew “B.” The training involved simulator demonstrations and exercises for feedwater failures and transients that could cause a reactor trip. The training department incorporated the exercises into Lesson Plan LRQ-SIM-D3, “Feedwater Malfunctions,” used for the training, and the abnormal conditions required operators to use Off-Normal Operating Procedure ONOP-FW-1, “Loss of Feedwater,” for recovery actions. The inspector also discussed several operator performance elements that the Crew “B” shift manager maintained for his crew to emphasize during simulator training.

b. Findings

There were no significant findings identified during this inspection

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspectors reviewed problems involving structures, systems, and components (SSCs) within the scope of the Maintenance Rule (10 Code of Federal Regulation (CFR) 50.65), as listed below, to evaluate the effectiveness of the licensee’s Maintenance Rule program. The reviews focused on proper maintenance rule scoping, proper classification of SSC equipment failures, safety significance and performance classifications described in, 10 CFR 50.65 (a)(1) and (a)(2), and performance criteria for SSCs classified as (a)(2). The inspectors reviewed the licensee’s applicable scoping documents, deficiency/event reports (DERs), and completed work orders related to the following SSC deficiencies:

- Central Control Room (CCR) Heating, Ventilation and Air Conditioning (HVAC)
  Several DERs relating to the CCR air conditioning unit 31 were reviewed: DERs 00-02719, 00-02853, 00-02857, 00-03000, 00-03020, 00-03034, 00-03146, 00-03206, and 00-03228.

  The DERs reviewed documented multiple trips of one of the two air compressors in the 31 air conditioning unit. The CCR HVAC is comprised to two air conditioning units (Nos. 31 and 32), each with two air compressors. The tripping of one air compressor reduced the effectiveness of the associated air conditioning unit but did not render it inoperable. In addition, the inspectors reviewed the Updated Safety Analysis Report (UFSAR) which stated that one air conditioning unit, along with minimizing central control room heat loads, would be sufficient to maintain the control room temperature below the maximum tolerable limit. The inspector further verified that System Operating Procedure, SOP-V-4, “Control Room Heating, Ventilation and Air Conditioning System,” Revision 11 required the operators to maintain the CCR temperature below the maximum limit. Therefore, the tripping of one air compressor in one air conditioning unit was not a functional failure of the CCR HVAC system, and no Maintenance Rule performance criteria were exceeded.

  In addition, the inspectors reviewed past DERs associated with the CCR HVAC system and observed a history of equipment problems with the air conditioning units and several attempts by the licensee to address these problems. The CCR air conditioning unit equipment problems remained a recurring issue throughout the inspection period.

- 33 Containment Fan Cooler Unit (FCU) Damper Failure, DER 00-03133

  During the performance of surveillance test, 3PT-Q77, “Containment Fan Cooling Units Manual Isolation Valves," on December 9, 2000, the 33 containment FCU damper "C" would not go to its normal position. The inspectors verified that damper "C" had failed to its required emergency position and the 33 FCU would have been capable of performing its emergency function if required. Therefore, this failure of damper "C" was not considered a functional failure of the 33 FCU, and no Maintenance Rule performance criteria were exceeded.

b. <u>Findings</u>

There were no significant findings identified during this inspection.

1R13 Maintenance Risk Assessment and Emergent Work

a. Inspection Scope (71111.13)

The inspectors reviewed the maintenance risk assessments and corrective maintenance work packages for the following emergent work, discussed the deficient conditions with cognizant personnel (system engineers, maintenance technicians, etc.), and observed the following work activities in the field:

- Work Request (WR) 94-00707-11; Replacement of the 34 Main Feedwater Regulating Valve Controller at Power. The licensee's risk assessment was also applicable for the replacement that was actually performed with the plant off-line.

- WR 00-01404-00; 32 Isophase Bus Duct Cooling Fan Repair

b. Findings

There were no significant findings identified during this inspection.

1R14 Personnel Performance During Non-Routine Plant Evolutions and Events

a. Inspection Scope (71111.14)

Forced Outage to Replace Hydrogen Coolers in the Main Turbine-Generator

On December 18, 2000, the licensee noted an increase in the makeup rate of hydrogen to the main turbine generator that was considered excessive, and initiated an unplanned shutdown to replace two hydrogen coolers (31 & 32). The 32 cooler was known to be leaking for several months. The licensee also decided to replace the 34 feedwater regulating valve (FRV) controller, and to repair the 32 isophase bus duct cooling fan during the outage. The inspector observed the licensee's planning and preparations for the outage, and observed operator performance in the control room during the shutdown. Operators completed a controlled shutdown without experiencing any significant equipment problems or system transients. However, the core quadrant power tilt ratio (QPTR) exceeded its alarm point of 102% at approximately 60% power, and continued to increase as power level decreased. Operators continued the power decrease at a controlled rate and obtained <2% power in less than two hours. The Technical Specifications limits on QPTR did not apply below 2% power, and consequently, an entry into a Limiting Condition for Operation (LCO) was not necessary for that condition. The QPTR continued to increase to a maximum of approximately 115% when reactor power reached 0-1%, and the licensee initiated actions for an exigent Technical Specification amendment to remove the restriction on QPTR below 50% power.

Plant Startup Following the Forced Outage to Replace Hydrogen Coolers

The inspectors observed operator action during the plant startup on December 20, 2000, following the forced outage to replace hydrogen coolers in the main generator. Prior to restarting the plant, the licensee obtained a Technical Specification (TS) amendment which stated that the limits on Quadrant Power Tilt Ratio (QPTR) limit were not applicable below 50% power. During power ascension, the QPTR exceeded the 1.02 limit when the plant reached 50% power and remained above the limit until approximately 90% power. In accordance with the TS amendment, the quadrant power tilt must be eliminated with 2 hours, or compensatory actions must be taken to restrict core power and to reset the nuclear instrument high flux trip setpoint. The operators increased power from 50% to 90% within 2 hours and eliminated the excess tilt. The licensee initiated DER 00-03265 to document ongoing QPTR issues during startups.

Partial Stroke Test of the 34 Feedwater Regulating Valve (FRV)

The inspector observed operator actions during the performance of the partial stroke test of the 34 FRV on December 21, 2000. This evolution was significant for the following reasons.

- There had been previous performance deficiencies with the 34 FRV controller known to the operators (Problem Identification PID #49322).
- The 34 FRV did not close fully during the plant shutdown on October 25, 2000.
- A stroke test had not previously been performed while the plant was at power.
- Two packing adjustments, which are known to have contributed to prior performance deficiencies, had been performed since the October 27, 2000, plant startup.

Failure of a Safety Injection (SI) System Logic Relay

During functional test 3PT-M14B of the safety injection system actuation logic on December 15, 2000, I&C technicians identified an open coil circuit in relay RTX-11. This failure would have prevented operators from manually resetting the safety injection (SI) logic following initiation of an SI signal, and also would have prevented a feedwater isolation following a reactor trip on low reactor coolant system average temperature (Tave). Consequently, operators entered the Limiting Condition for Operation (LCO) under Technical Specification Table 3.5-3. Note 6, which requires that the plant to be placed in hot shutdown within four hours of discovery of the failure.

The inspector observed control room activities following entry into the LCO. The shift manager and the control room supervisor initiated actions with the I&C department to obtain a replacement relay and to prepare for its installation. They also coordinated activities with the onshift crew to prepare for a rapid plant shutdown in the event that the relay could not be replaced within the allowed four hour LCO time limit. The control room supervisor coordinated an orderly review of the normal shutdown procedure with the auxiliary and control room operators, and established the time that the shutdown would have to be initiated so that hot shutdown would be achieved in one and one-half hours to meet the LCO time limit. I&C technicians obtained a replacement relay, conducted shop testing, installed the relay in the control room, and completed post-installation functional testing within approximately two hours. Operators subsequently exited the four hour LCO condition and averted further actions to perform a plant shutdown. DER 00-03186.

b. Findings

There were no significant findings identified during this inspection.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspector reviewed various DERs on degraded or non-conforming conditions that raised questions on equipment operability. The inspector reviewed the resulting operability determinations (ODs) for technical adequacy, whether or not continued operability was warranted, and to what extent other existing degraded systems adversely impacted the affected system or compensatory actions. The following DERs, calculations, and operability evaluations were evaluated:

- OD 00-33: During a plant shutdown on December 12, 2000, the 32 Main Steam Stop Valve failed to fully close following a manual trip of the main turbine and during subsequent testing. The inspector reviewed this problem with plant operators to evaluate the valve's condition. Plant personnel investigating the valve indicated that it was slightly offset in its seat, but that it would have fully closed following a automatic turbine trip at power, and that it would have closed sufficiently to isolate the turbine in the event that the main steam isolation valves had been disabled during a 10 CFR 50, Appendix R fire. DER 00-03242.

- Following a routine pressure relief of the plant's vapor containment on December 8, 2000, an operator noted an abnormally high flow in the weld channel and containment penetration pressurization (WCCPP) system. Flow was observed at approximately 1.9 standard cubic feet per minute (scfm), which was above the normal 1.0 scfm for the system. Operators isolated the leakage to Zone II of the WCCPP system which includes the 95 foot containment air lock. After cycling the outer air lock door open and closed, the system flow returned to normal. Two additional occurrences of high flow occurred in Zone II on December 23 and 28, and in both cases the flow was less than the alarm setpoint of 3.6 scfm and maximum system flow of 15 scfm. The licensee again reduced the WCCPP system leakage to normal and initiated a plan to replace to air lock door seal. DERs 00-03123, 00-03282, 00-03308.

b. Findings

There were no significant findings identified during this inspection.

1R17 Permanent Plant Modifications

a. Inspection Scope (71111.17A)

During an unplanned outage on December 19, 2000, the licensee performed a permanent modification of the 34 FRV controller, and replaced the existing Foxboro controller with an NUS model AMS700-AM. The modification was accomplished in accordance with design change 97-3-439. The existing controller was classified as quality assurance “Category 1" equipment, and the replacement controller was classified as ”non-category” (i.e., non-safety grade) . Accordingly, the design change involved installation of new safety grade signal isolators as boundary separation modules that disassociate the controller from other safety grade circuits and declassify the controller from “Category 1" equipment to “non-Cat 1” equipment in accordance with nuclear safety evaluation 97-03-439 MULTI. The inspector reviewed the design change documents listed above, discussed the details of the modification with plant personnel, and observed the equipment installation.

b. Findings

There were no significant findings identified during this inspection.

1R19 Post-Maintenance Testing

a. Inspection Scope (71111.19)

The inspectors reviewed post-maintenance test procedures and associated testing activities to assess whether 1) the effect of testing in the plant had been adequately addressed by control room personnel, 2) testing was adequate for maintenance performed, 3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing documents, 4) test instrumentation had current calibrations, range, and accuracy for the application, and 5) test equipment was removed following testing. The following surveillance activities were evaluated:

- WR 00-02633-01, post-maintenance test following air supply solenoid valve replacement and instrument air leak repair on the 33 Fan Cooler Unit, on December 13, 2000

  During the performance of surveillance test, 3PT-Q77, “Containment Fan Cooler Unit Manual Isolation Valves,” on December 9, 2000, the 33 containment FCU damper "C" would not go to its normal position. Initial troubleshooting which determined that the solenoid operated valve should be replaced was ineffective in correcting the malfunction and the 33 FCU damper "C" failed its post-maintenance test. The inspector verified that the licensee initiated a Deviation/Event Report to document the ineffective corrective action, DER 00-03156. The second troubleshooting and corrective action, repair of an instrument air leak in the damper piston, were successful in correcting the malfunction and the 33 FCU damper "C" passed its post-maintenance test.

- WR 94-00707-11; Post-installation test of the 34 Feedwater Regulating Valve Controller (FIC-447); and stroke time tests of the 32, 33, and 34 FRVs in accordance with test, ENG-623, “Feedwater Regulator Valves Timing,” following packing adjustments. All FRVs tested met their stroke time acceptance criteria specified in the procedure.

b. Findings

There were no significant findings identified during this inspection.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspector reviewed surveillance test procedures and associated testing activities to assess whether 1) the test preconditioned the component(s) tested, 2) the effect of testing on plant conditions was adequately addressed in the control room prior to and during testing, 3) the acceptance criteria demonstrated operational readiness consistent with design calculations and licensing documents, 4) the test equipment range and accuracy was adequate with proper calibration, and 5) the test was performed in the proper sequence by procedure.

The inspector reviewed and observed portions of the following surveillance tests and performed a review of related historical data and surveillance performance.

- 3PT-M79A; “31 Emergency Diesel Generator (EDG) Functional Test.” The inspector attended the prejob briefing for personnel involved in the test and observed engine operation inside the EDG cell. The inspector also observed maintenance support activities to assure the normal functioning of the engine fuel oil day tank inlet valves (SOV-1207A/B) after a recent failure to close properly.

- 3PT-M14B, “Safety Injection System Logic Functional Train B.”

b. Findings

There were no significant findings identified during this inspection.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

a. Inspection Scope (71151)

Heat Removal (Auxiliary Feedwater System) and High Head Safety Injection Systems Safety System Unavailability

The inspector reviewed the licensee's data supporting the performance indicators (PIs) for the auxiliary feedwater (AFW) and high head safety injection systems unavailability for the 4th quarter of 1999 and the 1st quarter of 2000. The inspectors reviewed the Deviation/Event Report (DERs), work request, and Limited Condition for Operations (LCOs) databases to identify equipment problems and system outages. In addition, the inspectors interviewed the performance engineer responsible for the data collection for these PIs.

b. Findings

There were no significant findings identified during this inspection.

4OA2 Identification and Resolution of Problems

a. Inspection Scope (71152)

During the performance of surveillance test, 3PT-Q97, "Steam Generator (SG) Level Channel Functional Surveillance Test," operators took actions contrary to their training and contrary to plant standard PS-04.03, "Conservative Decision Making." The operating crew made three attempts to place the 34 FRV controller into automatic control before being successful on the fourth attempt without requesting assistance from technical support groups or notifying operations management. In addition, during the fourth attempt, the operating crew intentionally created a small (~2%) deviation between steam and feedwater flow in order to counteract the FRV response observed during the first three attempts. While this small deviation was within the provisions of System Operation Procedure SOP-FW-001, "Main Feedwater System Operation," the training department had instructed operators to equalize the steam and feed flows prior to swapping from manual to automatic valve control.

The licensee identified the inappropriate actions as a human performance issue and critiqued the operating crew, however, the licensee did not initiate a Deviation/Event Report (DER) as required by administrative procedure AP-8, "Deviation & Event Report Initiation." AP-8 provided examples of adverse conditions including "...conditions detrimental to performance, including human performance deficiencies..." The inspector determined that the licensee, while appropriately identifying the human performance issue, was not effective in initiating a Deviation/Event Report. Inspector discussions with operations management were necessary to highlight the need to enter this item into the corrective action process for resolution and trending. This human performance issue will be tracked under DER 00-03316.

b. Findings

There were no significant findings identified during this inspection.

4OA4 Licensee Event Report Reviews

(Closed) Licensee Event Report (LER) 2000-11-00 Reactor Core Quadrant Power Tilt Ratio Exceeded Technical Specification Limit During Startup and Specified Actions Not Taken; A Condition Prohibited by Technical Specifications

On November 27, 2000, the licensee submitted LER 2000-011-00 to the NRC documenting that the quadrant power tilt ratio exceeded the technical specification (TS) limit during startup on October 27, 2000 and the required actions were not taken within the specified time. The event documented in this LER was discussed in Section 4OA7, Licensee Identified Violations, of Inspection Report (IR) 2000-007, documented as a non-cited violation (NCV 05000286/2000-007-01), and entered into the corrective action system (DER 00-02781). The event report satisfied the requirements of 10CFR50.73, and no new issues were revealed by this LER. This LER is closed.

4OA6 Meetings

Exit Meeting Summary

On January 11, 2001, the inspectors presented the inspection results to Mr. F. Dacimo and other Entergy staff members who acknowledged the inspection results presented. The inspector asked Entergy personnel whether any materials evaluated during the inspection were considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

R. Barrett	Site Vice President
J. Barry	Sr. Radiological Engineer
R. Burroni	I&C Manager
F. Dacimo	Plant Manager
J. Comiotes	General Manager-Operations
J. DeRoy	Director, IP-3 Engineering
R. Deschamps	Radiological and Environmental Services Manager
M. Dinelli	Performance Engineer
Z. Eisenberg	System Engineer
D. Mayer	General Manager-Support Services
J. Perrotta	Quality Assurance Manager
K. Peters	Licensing Manager
P. Rubin	Operations Manager
J. Russell	General Manager-Maintenance
I. Sinert	System Engineer
B. Sullivan	Assistant Operations Manager
S. Van Buren	Fire Protection Engineer
A. Vitali	Maintenance Manager
J. Wheeler	Training Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Closed</u>

LER 2000-11-00	Reactor Core Quadrant Power Tilt Ratio Exceeded Technical Specification Limit During Startup and Specified Actions Not Taken; A Condition Prohibited by Technical Specifications

LIST OF ACRONYMS USED

AFW	auxiliary feedwater
CCR	central control room
CCW	component cooling water
CFR	Code of Federal Regulations
COL	checkoff list
DER	Deviation/Event Report
FCU	fan cooling units
FRV	feedwater regulating valve
HVAC	heating, ventilation, and air conditioning
IR	inspection report
LCO	Limiting Condition for Operation
LER	Licensee Event Report
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
OA	Other Activities
OD	operability determination
ONOP	Off-Normal Operating Procedure
PARS	Publicly Available Records
PI	performance indicator
PID	problem identification
QPTR	quadrant power tilt
scfm	standard feet per cubic minute
SCBA	self-contained breather apparatus
SCFM	Standard Cubic Feet Per Minute
SI	safety injection
SOP	system operating procedure
SSCs	structures, systems and components
Tave	reactor coolant system average temperature
TS	Technical Specification
UFSAR	Update Final Safety Analysis Report
WCCPP	Weld Channel and Containment Penetration Pressurization
WR	work request

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

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	**Radiation Safety**	**Safeguards**
• Initiating Events	• Occupational	• Physical Protection
• Mitigating Systems	• Public	
• Barrier Integrity		
• Emergency Preparedness		

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 margins and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margins 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.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.