UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

January 29, 2001

SDP/EA-00-137

Duke Energy Corporation
ATTN: Mr. W. R. McCollum
Site Vice President
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 50-269/00-07, 50-270/00-07, AND 50-287/00-07

Dear Mr. McCollum:

On December 30, 2000, the NRC completed inspections at your Oconee facility. The enclosed report documents the inspection findings which were discussed on January 9, 2001, with Mr. M. Nazar and other members of your staff.

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

Based on the results of this inspection, the inspectors identified four issues of very low safety significance (Green). Three of these issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny the non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Oconee facility.

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 NRC's document system

(ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADMAS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Robert Haag, Chief
Reactor Projects Branch 1
Division of Reactor Projects

Docket Nos: 50-269, 50-270, 50-287
License Nos: DPR-38, DPR-47, DPR-55

Enclosure: NRC Integrated Inspection Report 50-269,270,287/00-07, w/Attached NRC's Revised Reactor Oversight Process

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-269, 50-270, 50-287
License Nos:	DPR-38, DPR-47, DPR-55
Report No:	50-269/00-07, 50-270/00-07, 50-287/00-07
Licensee:	Duke Energy Corporation
Facility:	Oconee Nuclear Station, Units 1, 2, and 3
Location:	7800 Rochester Highway Seneca, SC 29672
Dates:	September 31 - December 30, 2000
Inspectors:	M. Shannon, Senior Resident Inspector D. Billings, Resident Inspector E. Chrisnot, Resident Inspector S. Freeman, Resident Inspector E. Testa, Radiation Protection Inspector (Sections 2OS1, 2OS2, and 2PS2)
Approved by:	R. Haag, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000269,270,287/00-07, on 09/31 - 12/30/2000, Duke Energy Corporation, Oconee Nuclear Station, Units 1, 2, and 3. Adverse weather protection, maintenance risk assessments and emergent work evaluations, post maintenance testing, and surveillance testing.

The inspection was conducted by resident inspectors and a regional radiation protection inspector. The inspection identified four Green findings, three of which involved non-cited violations (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using the Significance Determination Process (SDP) found in Inspection Manual Chapter 0609. Findings to which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation.

Cornerstone: Mitigating Systems

- Green. A non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, was identified for failure to implement timely corrective actions following freezing of a borated water storage tanks (BWST) level sensing line in 1996. A failure of both heat trace circuits with non-functioning alarms allowed this condition to occur. The licensee has not implemented the identified corrective actions to reactivate the heat trace alarm circuits for BWST level sensing lines.

  Because no BWST level instrument sensing lines have frozen since the 1996 occurrence and the heat trace circuits for the BWST level instruments were operating, the inspectors determined that this issue was of very low safety significance (Section 1R01).

- Green. A finding was identified in that the licensee did not develop written contingency plans prior to removing the Keowee hydro unit (KHU) underground feeder from service. This placed Unit 2 and Unit 3 in an orange Oram-Sentinal condition. Procedural guidance directed that written contingency plans be developed.

  The issue was determined to be of very low safety significance based on the licensee's determination that the condition (KHU underground feeder being inoperable) was not an actual orange Oram-Sentinal condition (Section 1R13.2).

- Green. A non-cited violation of Technical Specification (TS) 3.3.1 was identified for the failure to maintain three channels of the reactor protection system operable for the turbine trip and loss of main feedwater functions, in that the as-left setpoints from previous calibrations did not meet the allowable values specified in TS Table 3.3.1-1.

  Because the setpoints only slightly exceeded the TS allowable values, this issue was of very low safety significance (Section 1R22.2).

Cornerstone: Barrier Integrity

- Green. A non-cited violation of 10 CFR 50, Appendix B, Criterion XVII, was identified for failure to maintain sufficient records to furnish evidence of activities affecting quality. The licensee failed to document the results on the Penetration 19 Leak Rate data sheets following failure of the containment purge valves to pass the leak rate test.

This issue had very low safety significance since the lack of test failure documentation did not affect repair of the purge valves and subsequent successful testing (Section 1R19.2).

Report Details

Summary of Plant Status:

Unit 1 began the period at 100 percent power and remained at that power level except for brief periods of power reduction for control rod and main turbine valve testing. On November 23, 2000, the unit shutdown for the End of Cycle (EOC) 19 refueling outage. The unit remained in the EOC-19 outage for the remainder of the inspection period.

Unit 2 was at 100 percent power throughout the inspection period except for brief periods of power reduction for control rod and main turbine valve testing. On December 19, 2000, the unit entered TS 3.0.3 due to both A and B control room chillers being inoperable for approximately two and one-half hours. The unit power was reduced to approximately 99 percent full power for one hour and when the TS was exited the unit power was returned to 100 percent.

Unit 3 was at 100 percent power throughout the inspection period except for brief periods of power reduction for control rod and main turbine valve testing. On December 19, 2000, the unit entered TS 3.0.3 due to both A and B control room chillers being inoperable for approximately two and one-half hours. The unit power was reduced to approximately 99 percent full power for one hour and when the TS was exited, the unit power was returned to 100 percent.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee’s preparations to protect the Unit 1, 2, and 3 BWSTs and associated piping from freezing during cold weather. The review included: the design of the heat trace system, the procedures used to check operation of the heat trace circuits, a sample of work orders used for the most recent implementation of that procedure, and Problem Investigation Process (PIP) documents. The inspectors also walked down the BWSTs, the associated piping and heat trace, the heat trace breaker/alarm panels, and the heat trace power supplies. During cold weather the inspectors took surface temperature measurements of the heat traced pipes at each BWST to verify heat tracing was properly functioning. Specific documents reviewed included:

- IP/0/B/1606/009, Preventive Maintenance And Operational Check of Freeze Protection, Revisions 9 & 11.
- OP/1/A/1102/020, Revision 92, Enclosure 5.8, Cold Weather Checklist.
- WO 98283631-01, Unit 1 Freeze Protection PM.
- WO 98283631-02, Unit 2 Freeze Protection PM.
- WO 98283631-03, Unit 3 Freeze Protection PM.
- PIPs: O-96-00252, O-96-00285, O-96-00639, O-96-02805, O-98-00001, O-98-01103, O-98-03118, O-99-00376, O-99-04632, O-00-00319.

b. Findings

A Green finding was identified for failure to implement timely corrective actions following the freezing of a BWST level sensing line in 1996 and was dispositioned as an NCV. Specific corrective actions not completed dealt with reinstalling alarms to alert operators of heat trace failures. This issue was considered to be of low safety significance because none of the BWST level sensing lines have frozen since the 1996 occurrence.

The original BWST heat trace circuits for all three units were designed with alarm circuits in cabinets located on the first floor of the auxiliary building. Each cabinet contained a power supply breaker, current transducer (CT), and alarm for each circuit. Individual circuit alarms actuate when the thermostat drops below its setpoint and the CT fails to detect current. However, in 1989, when the licensee modified the heat trace circuits on the BWST level instrument sensing lines, the alarm circuits were removed. This was caused by the new thermostats (different manufacturer) not containing the auxiliary contacts for the alarm circuit. The inspectors noted that following the modification of the heat trace circuitry, the licensee did not implement any compensatory measures to verify continued operation of the BWST level instrument sensing line heat trace circuits or to periodically check the operation of the BWST heat trace circuits. Subsequently, in 1996, following failure of the BWST level sensing line heat trace circuits and freezing of the sensing lines, the licensee implemented Procedure IP/0/B/1606/009 which performed an annual check of the heat trace circuits, typically during the summer. Based on the present non-functional alarm circuits, the inspectors concluded that operators would not be aware of heat trace circuit failure on the BWST level instruments until after an instrument sensing line froze. During the inspection, the inspectors measured the surface temperature of all accessible heat traced pipes on each BWST during freezing conditions and determined that the heat trace circuits were operating.

The inspectors noted several problems with procedure IP/0/B/1606/009 which was used to check the operation of heat trace circuits used to provide freeze protection. For example, the procedure did not include a functional test of the heat trace alarm circuits, the procedure did not provide acceptance criteria or tolerances for operation of the thermostats, the procedure did not provide a list of setpoints or tolerances for the thermostats and strip heaters in the BWST instrument boxes, and the procedure called for a current measurement on each heat trace circuit and provided a generic acceptance criteria of +/- 10 percent of the amperage listed which was not was not based on any standard or manufacturer's recommendation.

During a review of the corrective action documents associated with the heat trace system, the inspectors noted that on February 5, 1996, the sensing lines for a Unit 2 BWST level instrument became frozen when both heat trace circuits failed, thus making the level instrument inoperable. At that time the licensee confirmed that the heat trace circuits for the BWST level instruments contained no alarms. Subsequently, the licensee proposed a corrective action to modify the thermostats to reestablish the heat trace circuit failure alarms. However, the heat trace circuitry corrective actions have been deferred until 2003 as part of the Oconee Refurbishment Program. Based on the safety significance of the BWST level instruments, the inspectors concluded that the licensee had not implemented timely corrective action to resolve this issue.

The inspectors noted other problems with the identification and resolution of heat trace equipment malfunctions. For example, when procedure IP/0/B/1606/009 was performed on August 14, 2000, and October 12, 2000, the as-found setpoints for one thermostat did not match the design setpoint. No PIP was initiated for the problem even though the affected circuit was required by Selected Licensee Commitment (SLC) 16.5.13. During the same performance of procedure IP/0/B/1606/009, the as-found current readings on several circuits were outside the 10 percent acceptance criteria specified in the procedure. No PIP was initiated on this problem and no corrective actions were taken. On November 13, 2000, during a transfer from the Unit 1boric acid mixing tank (BAMT) to the Unit 1 Concentrated Boric Acid Storage Tank (CBAST), the inspectors noted that a CBAST heat trace circuit required by SLC 16.5.13 went into alarm. At that time, the operators explained that the alarm occurred regularly during such transfers and improperly concluded that this was not a problem. However, when the inspectors questioned how the alarm circuit actually functioned, it was determined that the alarm indicated a circuit failure and the licensee subsequently found a blown fuse in the circuit. This alarming failed circuit had not been identified in a PIP. These problems were later entered into PIP O-00-04377.

The failure to take prompt corrective action for the 1996 freezing of BWST level instrument sensing lines, i.e., restore heat trace alarms, is considered to have a credible impact on safety. The inspectors noted that the operators would not be aware of heat trace circuit failure until after a BWST level instrument sensing line would actually become frozen, making that instrument inoperable. Because no BWST level instrument sensing lines have frozen since the 1996 occurrence and the heat trace circuits at the BWST were operating, the inspectors determined that this issue was of very low safety significance (Green).

The freezing of the BWST level sensing instrument in 1996 was considered to be a condition adverse to quality. The failure to reactivate the BWST heat trace alarm circuits within a reasonable time or to develop contingency actions to monitor the BWST level sensing line heat trace circuits during cold weather until the heat trace alarm circuits could be reactivated was considered to be untimely corrective actions. The inspectors considered this failure to promptly identify and correct a condition adverse to quality as a violation of 10 CFR 50 Appendix B, Criterion XVI. This violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 50-269,270,287/00-07-01: Inadequate Corrective Actions on BWST Level Instrument Heat Trace. This violation is in the licensee’s corrective action program as PIP O-00-04377.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted partial equipment alignment walkdowns, of those systems identified below, to evaluate the operability of selected redundant trains or backup systems. The walkdowns included, as appropriate, reviews of plant procedures and other documents to determine correct system lineups, and verification of critical components to identify any discrepancies which could affect operability of the redundant train or backup system.

- The Unit 2B penetration room ventilation system during maintenance of the Unit 2A penetration room ventilation system (October 16, 2000)
- Emergency power system, standby bus, and main feeder bus alignment to Units 2 and 3 during outage work on the Unit 1 electrical switchgear (inoperable Keowee underground path) (December 3, 2000)
- Decay heat removal system, borated water storage tank gravity feed to the reactor vessel, bleed hold up tank pumping system, normal and emergency electrical power systems, and the in-core temperature monitoring system during reduced inventory activities (November 27-28, 2000)

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours of selected areas to verify that combustibles and ignition sources were properly controlled, and that fire detection and suppression capabilities were intact. The inspectors selected the areas based on a review of the licensee's safe shutdown analysis and the probabilistic risk assessment based sensitivity studies for fire-related core damage accident sequences. Inspection of the following areas were conducted during this inspection period: the unit 3 control room ventilation room, cable spreading room, and east and west penetration rooms; the unit 1 and unit 2 cable spreading rooms and the east and west penetration rooms; and both Keowee units.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the low pressure injection/building spray (LPI/BS) room cooling heat exchanger preventive maintenance and testing to ensure that the room coolers would be able to supply the necessary cooling as described in the Final Safety Analysis Report (FSAR). The inspection focused on deficiencies that could mask degraded performance of the heat exchangers, could result in common cause heat sink performance problems, and ensure that the licensee has adequately identified and resolved heat sink performance problems that could affect multiple heat exchangers in mitigating systems. The inspectors walked down the system, reviewed the preventive maintenance activities associated with the LPI/BS room coolers, and reviewed the surveillance activities associated with the LPI/BS room coolers.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed simulator training, on November 13, 2000, for reactor operators and senior reactor operators. The inspectors observed a loss of power scenario with a subsequent small break loss of coolant accident which developed into a moderate break coolant accident. The inspectors evaluated crew's performance in terms of communications; ability to take timely actions in the safe direction; prioritizing, interpreting, and verifying alarms; correct use and implementing of procedures, including the alarm response procedures; timely control board operation and manipulation, including high-risk operator actions; and oversight and direction provided by the shift supervisor, including the ability to identify and implement appropriate TS actions, such as reporting, emergency plan actions, and notifications. The inspectors also attended the evaluators critique.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors sampled portions of selected structures, systems, and components (SSCs) listed below, as a result of performance-based problems, to assess the effectiveness of maintenance efforts that apply to scoped SSCs. Reviews focused, as appropriate, on: (1) maintenance rule scoping in accordance with 10 CFR 50.65; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 50.65 (a)(1) or (a)(2) classifications; and (5) the appropriateness of performance criteria for SSCs classified as (a)(2) or goals and corrective actions for SSCs classified as (a)(1). The selected SSCs were as follows:

- Reactor Vessel Level Transmitters LT-5A and LT-5B
- Vital Station Batteries
- Reactor Building Cooling Unit 1B
- Lee Combustion Turbine 5C

- Unit 1 LPI Valves 1LP-9 and 1LP-10
- Auxiliary Building Ventilation LPI Room Cooling Strainers

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluations

.1 Assessments and Evaluations

a. Inspection Scope

The inspectors evaluated, as appropriate for the selected SSCs listed below: (1) the effectiveness of the risk assessments performed before maintenance activities were conducted; (2) the management of risk; (3) that, upon identification of an unforseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and (4) that maintenance risk assessments and emergent work problems were adequately identified and resolved. The following items were reviewed under this inspection procedure:

- Spent Fuel Pool Carriage maintenance
- Reactor Protection System Turbine Trip and Feedwater Pump Trip Setpoint(s) Recalibration(s)
- Emergency Siphon Seal Water Pump 2A maintenance
- Isolation of elevated water storage tank (fire system head tank)
- WO 98330672, Failure of the Keowee Unit 2 partial shut solenoid 99SN, as documented in PIP O-00-3920
- Emergency power system testing on Unit 1 affecting the operating Units 2 and 3
- Testing of the Lee Power Station combustion turbines for supply of the standby busses

b. Findings

No findings of significance were identified.

.2 Emergency Power System Risk Assessment

a. Inspection Scope

The inspectors observed and reviewed the risk assessment of the emergency power system, standby bus, and main feeder bus to Units 2 and 3 during outage work on the Unit 1 electrical switchgear.

b. Issues and Finding

A Green finding was identified for the failure to develop contingency plans prior to taking the KHU underground feeder out of service. Following the licensee's determination that the condition (KHU underground feeder being inoperable) was not an actual orange Oram-Sentinal condition, this issue was determined to be of very low safety significance.

During the Unit 1 refueling outage, on December 3, 2000, the 4160 volt switchgear 1TC was removed from service for planned maintenance. The KHU underground feeder was made inoperable due to disconnecting the 4160 volt underground supply from switchgear 1TC. The equipment configuration placed the shut down unit (Unit 1) in a green defense in depth assessment condition. However, this also resulted in the operating units (Unit 2 and Unit 3) being placed in an orange Oram-Sentinel risk condition. Based on the licensee's risk management program, when an operating unit is going to be placed in an orange Oram-Sentinel risk condition, a written contingency plan must be developed. As part of the review for risk management actions the inspectors requested a copy of the contingency plan. A copy of the written contingency plan was not available in that the work control center did not have a copy and the Unit 2 control room did not have a copy. The inspectors were later informed that a plan had not been developed prior to the start of the planned maintenance. Subsequently, on December 5, 2000, the 1TC switchgear was again being placed in an inoperable status for surveillance testing. Prior to the testing activities, operations personnel stopped the activities until the written contingency plans were in place.

The issue was determined to be of very low safety significance (Green) by the significance determination process. However, the licensee's failure to develop contingency plans for a true orange risk condition would be a more significant safety concern, in that, actions to compensate for an increased risk condition could not be taken. This issue is in the licensee's corrective action as PIP O-00-4384.

1R14 Personnel Performance During Nonroutine Plant Evolutions

a. Inspection Scope

The inspectors reviewed personnel performance during selected non-routine events and/or transient operations. As appropriate, the inspectors: (1) reviewed operator logs, plant computer data, or strip charts to determine what occurred and how the operators responded; (2) determined if operator responses were in accordance with the response required by procedures and training; (3) evaluated the occurrence and subsequent personnel response using the SDP; and (4) confirmed that personnel performance deficiencies were captured in the licensee's corrective action program. The non-routine evolutions reviewed during this inspection period included the following:

- TS 3.0.3 Entry Due to Multiple Channels of the Reactor Protection System Being Inoperable Due to Improper Calibration of Trip Setpoints on October 11, 2000
- Minor Turbine Building Flooding with Unexpected Leakage into the Auxiliary Building on December 13, 2000
- TS 3.0.3 Entry During Testing of the Low Pressure Service Water System due to Low Reactor Building Cooling Unit Flow on December 28, 2000

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations affecting the risk significant mitigating systems listed below, to assess, as appropriate: (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered; (4) if compensatory measures were involved, whether the compensatory measures were in place, would work as intended, and were appropriately controlled; and (5) where continued operability was considered unjustified, the impact on TS limiting conditions for operations (LCO). The inspectors reviewed the operability evaluations described in the following PIPs:

- PIP O-00-03552, Unit 3 high pressure injection pump test data re-analysis
- PIP O-00-03909, Reactor Building Cooling Unit 1B exhibited high stator temperatures in slow speed
- PIP O-00-04141, Unit 3 pressurizer relief valve seat leakage increase
- PIP O-00-03405, resolution of imbalance between the in-core and ex-core neutron detectors
- PIP O-00-04643, control room chill water unit trip due to air binding of the low pressure service water cooling supply to the chiller
- PIP O-00-03909, Reactor Building Cooling Unit 1B motor high temperature root cause failure analysis report

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

.1 Routine Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance test (PMT) procedures and/or test activities, as appropriate, for selected risk significant mitigating systems to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and/or engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy consistent with the application; (5) tests were performed as written with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; (7) test equipment was removed following testing; and (8) equipment was returned to the status required to perform its safety function. The inspectors observed testing and/or reviewed the results of the following tests:

- MP/0/A/1720/016, VT-2 Pressure Testing of the 10 inch and 3 inch LPI piping replaced by modifications ONOE-14877 and ONOE-14878, Revision 23
- PT/1/A/0152/012, LPI System Valve Stroke Test, Revision 14 following replacement of 1LP-17 and 1LP-18
- TT/1/A/0150/055, 1LP-17 and 1LP-18 Flow Test, Revision 0
- TT/0/A/0620/46, Keowee Recommended Start-up Area Curve Test (Overshoot), Revision 0
- TT/0/A/0620/48, Keowee Emergency Start After Governor Modification, Revision 1

b. Findings

No findings of significance were identified.

.2 Unit 1 Purge Valve Local Leak Rate Testing (LLRT)

a. Inspection Scope

The inspectors observed the testing and reviewed the results of the LLRT for Unit 1 containment purge system valves 1PR-5, 6, 29 and 30 related to containment penetration 19.

b. Observations and Findings

A Green finding was identified for the failure to document the results of containment penetration 19 (containment purge valves) leak rate testing for the test failure on December 29, 2000, as required by 10 CFR 50, Appendix B, Criterion XVII. This finding was dispositioned as an NCV. This issue had very low safety significance since the lack of test failure documentation did not affect the purge valve repairs and subsequent successful testing.

On December 29, 2000, the inspectors observed portions of procedure PT/1/A/0151/19, Penetration 19 Leak Rate Testing, Revision 6. The inspectors observed that step 12.3.1, titled, Ensure leak rate monitor setup; Substep 12.3.1.B, stated, regulate to 60 psig; and substep 12.3.8.A, stated, pressurize to leak rate monitor test pressure. The inspectors observed that substep 12.3.8.A could not be completed. This was due to the penetration only being able to be pressurized to approximately 40 psig. The test was stopped and the inspectors were informed that adjustments would have to be made to the valve seats. Subsequently, the inspector requested a copy of the completed test procedure to ensure that the results (obtained pressure and leak rate) of the failed test had been documented. The licensee could only provide the completed test results following valve maintenance, which indicated satisfactory completion of the test but did not document the test failure nor the results of the failed test. The inspectors noted that Enclosure 13.2 of the procedure, for “Leak Rate Test Failure Record” had been initialed and documented as not applicable (N/A).

This issue was determined to be of very low significance (Green) because the lack of documentation for the failed test did not adversely affect repairs and subsequent satisfactory testing of the purge valves. However, the inspector concluded that this issue could have a credible impact on safety, in that, by not documenting test failure data and actions taken to correct the deficiency, the licensee would not be able to identify degradations of the purge valves and the resulting impact on containment closure during mid-loop operations. A similar issue occurred when Unit 1 containment purge valve test failed due to excessive leakage which prevented the establishment of the test pressure. Valve repairs were required. This condition with excessive leakage was not documented as part of the testing records.

10 CFR 50, Appendix B, Criterion XVII, Quality Assurance Records, requires, in part, that sufficient records shall be maintained to furnish evidence of activities affecting quality and that inspection and test records shall, as a minimum, identify the results of testing, the acceptability of the test, and action taken in connection with any deficiencies noted. It also requires that “Records shall be identifiable and retrievable.” In addition, the licensee’s Nuclear System Directive, NSD 704, Technical Procedure Use and Adherence, Rev 8, Section 704.7.8.C, requires, in part, that a test deficiency be documented on the appropriate form within the procedure. The failure to document the test results on the Penetration 19 Leak rate data sheet, Enclosure 13.1, page 2, and failure to document the test failure on the leak rate test failure record, Enclosure 13.2, was considered a violation. This violation is being treated as an NCV, consistent with Section VI.A.1 of the enforcement policy and is identified as NCV 50-269/00-07-02: Failure to Document the Results of a Failed Test and Failure to Document a Test Failure. This violation is in the licensee’s corrective action program as PIP O-01-00364.

1R20 Unit 1 Refueling Outage

.1 Routine Unit 1 Refueling Outage Observations

a. Inspection Scope

The inspectors observed selected activities and reviewed associated documentation related to the Unit 1 refueling outage to verify conformance to applicable procedures. Surveillance tests were reviewed to ascertain completeness within the TS required specifications. The inspectors also reviewed the Duke Power Company Assessment Report, 1EOC19 Pre-Outage Risk Assessment, to determine if the Unit 1 refueling plan was done in accordance with applicable shutdown risk guidance. Activities observed included the following:

- reactor shutdown
- reactor cooldown and initiation of decay heat removal (DHR)
- calibration and operation of the low temperature overpressure (LTOP) reactor protective function
- reduced inventory RCS level instrument calibrations and operation
- reduced inventory and mid-loop conditions to install and remove steam generator nozzle dams
- fuel off loading and refueling operations
- electrical power alignments and testing during major outage activities
- reactor vessel head repair activities associated with one control rod drive mechanism and eight thermocouple nozzles
- containment closure
- outage-related surveillance tests (PT/1/A/0610/01L, Load Shed Channel Verification, Rev. 3, and PT/1/A/0610/01A, EPSL Functional Test, Rev. 22)

b. Findings

Except as follows, no findings of significance were identified.

.2 Unit 1 Containment Emergency Airlock Temporary Cover Plate Requirements

a. Inspection Scope

The inspectors reviewed the licensee’s commitments from Generic Letter (GL) 88-17 as stated in FSAR, Chapter 16, SLC 16.5.3. These commitments included the controls and administrative procedures governing mid-loop (reduced inventory) operation and containment closure capability.

b. Findings

An additional issue was identified for review along with previously identified unresolved item. This issue involved inadequate controls to respond to loss of DHR events and the resulting failure to implement the commitments to GL 88-17 and SLC 16.5.3 for ensuring or obtaining containment closure during reduced inventory operation of the reactor coolant system with fuel in the reactor. The inspectors observed that the licensee was using an aluminum plate on the emergency personnel access hatch to meet containment closure. The aluminum plate had not been designed to shutdown accident conditions.

Background

GL 88-17, Attachment 2, Guidance for Meeting Generic Letter, Section 2.2, Containment Closure, requested the licensee to "implement procedures and controls to assure that containment closure will be achieved prior to the time at which a core uncovery could result from a loss of DHR. Reasonable assurance of containment closure should include consideration of activities which must be conducted in a harsh environment. SLC 16.5.3, Loss of DHR, delineates the commitment to identify and ensure all containment penetrations would be identified and closed within a 2.5 hour period in the event of a loss of DHR. The basis for these actions is the GL 88-17 guidance.

During the conversion from Custom TS to Improved TS in 1999, the licensee submitted License Amendment Request (LAR) 99-03 that requested that a note be added to TS 3.9.3b to allow the use of the temporary cover plate for the emergency air lock during core alteration or movement of irradiated fuel within the containment. Attachment 3, Technical Justification stated, "In the event of a loss of DHR capability, personnel are designated by procedure to disconnect the temporary hoses, tubing and cabling and close the emergency air lock outer door to restrict potential leakage." It also stated, "The closure restrictions, including the allowance for the temporary cover plate, are sufficient to restrict fission product radioactivity release due to a fuel handling accident."

Problem Assessment

The licensee has determined that the temporary cover on the inner emergency personnel hatch opening meets the requirements for containment closure, including containment closure as discussed in GL 88-17. Attachment 2 of GL 88-17 stated that reasonable assurance of containment closure should include consideration of activities which must be conducted in a harsh environment. However, because the temporary cover on the emergency personnel hatch was considered sufficient to meet containment closure requirements, the inspectors noted that the applicable portion of the abnormal procedure (AP) which directs personnel to close the outer emergency personnel hatch would not be implemented.

The inspectors reviewed, AP/1/A/1700/026, Loss of Decay Heat Removal, Rev.10. Step 4.3 stated, "IF all of the following are true: core cooling does NOT exist AND Containment closure does NOT exist, THEN establish Reactor Building containment closure; refer to Attachment 6.1, Establishing Reactor Building Containment Closure and OP/1/A/1502/009, Containment Closure Control." Attachment 6.1 directs the Senior Reactor Operator to notify mechanical maintenance to close the Reactor building equipment hatch, all temporary Reactor building penetrations, and all SG secondary side penetrations inside containment. It further directs an operator to be dispatched to close at least one door in the Reactor building personnel and emergency hatches. However, since the licensee considers the temporary cover plate on the emergency personnel hatch to meet containment closure requirements, personnel would not be dispatched to close the outer door on the emergency personnel hatch. The inspectors interviewed several operations personnel on the meaning of containment closure related to the temporary cover plate. All personnel interviewed stated that the temporary cover plate met the requirements for containment closure.

Based on this, the inspectors concluded that the abnormal procedure in conjunction with the licensee's view on the aluminum cover failed to adequately implement the SLC requirements to ensure that containment closure be established on a loss of DHR Specifically that personnel would not be dispatched to remove the cables from the temporary cover plate and close an emergency hatch door prior to boiling in the core and possible pressurization of the reactor building.

Risk Significance

During the current Unit 1 Refueling Outage, Oconee had entered a reduced inventory condition from 1:21 p.m., November 27 until 8:07 a.m., November 28, 2000; and from 9:35 p.m., December 20 until 10:38 a.m., December 21, 2000. This was done to install and then remove nozzle dams for steam generator inspections. These reduced inventory conditions were completed with fuel in the vessel and reduced water levels that resulted in calculated times to boil of 18 minutes and 64 minutes respectively.

In a mid-loop drained condition a loss of DHR could result in a core melt and pressurization of the containment. The approximately 30 inch diameter aluminum temporary cover plate has not been analyzed nor was it designed to withstand a harsh environment or reactor building pressurization. A failure of this cover plate would result in a significant opening from containment atmosphere to the outside and potentially result in a major radioactivity releases in excess of 10 CFR 100 limits. In addition, following a potential failure of the temporary cover, the operators would not be able to close the outer emergency equipment hatch without making an entry into the emergency access penetration to disconnect the temporary cables and piping and then closing the outer hatch against any containment pressure.

The NRC is reviewing a similar issue dealing with containment purge valves which had excessive leakage when tested and the resulting consequences on mid-loop operations. This similar issue was previously identified as URI 50-269/00-05-11. As part of this URI review, the NRC will assess the significance of not having containment closure for loss of decay heat removal events while operating in mid-loop conditions. Since the issue with the aluminum cover not providing adequate containment closure capability involves the same type significance assessment, this issue is identified as another example to be reviewed under URI 50-269/00-05-11.

1R22 Surveillance Testing

.1 Routine Surveillance Testing Observations

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of the selected risk-significant SSCs listed below, to assess, as appropriate, whether the SSCs met TS, UFSAR, and licensee procedure requirements. In addition, the inspectors determined if the testing effectively demonstrated that the SSCs were ready and capable of performing their intended safety functions. The following testing was observed and/or reviewed:

- IP/0/A/0305/005D, Reactor Building Hi Pressure Trip, Revision 27
- PT/O/A/0400/011, Safe Shutdown Facility Diesel Generator Test, Revision 9
- PT/1/A/0610/01L, Load Shed Channel Verification, Revision 3
- TT/0/A/0261/07, CCW Pump Flange Test, Revision 3, a special test to determine if air in-leakage exists in the pump flange during low lake levels - commitment to the NRC
- PT/1/A/0610/01J, Emergency Power Switching Logic Functional Test, Revision 32

b. Findings

No findings of significance were identified.

.2 Calibration of RPS Anticipatory Trips

a. Inspection Scope

The inspectors reviewed the calibration of the RPS anticipatory trips from the main turbine and main feedwater systems. This included a review of Procedure IP/0/A/0305/011, RPS Channel C Main Feedwater Pumps and Main Turbine Trips Calibration, Revision 31; observing the performance of the procedure on Unit 1 RPS Channel C; and a review of PIP O-00-03607.

b. Findings

A Green finding was identified for failure to maintain three channels of RPS operable for the turbine trip and loss of main feedwater functions as required TS 3.3.1 and was dispositioned as an NCV. The as-left setpoints from previous calibrations did not meet the allowable values specified in Table 3.3.1-1. Because the setpoints only slightly exceeded the TS allowable values, the inspectors determined that this issue was of very low safety significance.

On October 11, 2000, the inspectors observed the calibration of the anticipatory trips from the main turbine and main feedwater systems on Unit 1 RPS Channel C. Procedure IP/0/A/0305/011 specified a setpoint of 800 psig +/- 10 psig for the main turbine trip, 75 psig +/- 5 psig for the main feedwater trip, and 2.4 volts DC (VDC) +/- 0.018 VDC for the turbine trip bypass. The inspectors compared these to the setpoints in TS Table 3.3.1-1 which have allowable values of greater than or equal to 800 psig for the main turbine; greater than or equal to 75 psig for main feedwater; and greater than or equal to 30 percent (2.4 VDC) rated thermal power for the turbine trip bypass. Therefore, the procedure allowed the setpoints to be set below the TS allowable value. The inspectors discussed this with the licensee who subsequently reviewed data from the most recent calibrations on all three units which had been performed in October 1999 for Unit 1, November 1999 for Unit 2, and March 2000 for Unit 3. The licensee determined that the as-left setpoints for the main turbine anticipatory trip exceeded the TS allowable values on Unit 1 RPS Channels A and C and Unit 3 Channels A and D. The licensee also found that the as-left setpoints for the main feedwater exceeded the TS allowable values on Unit 1 RPS Channels A, B, C, & D; Unit 2 Channel C; and Unit 3 Channels B, C, & D. Additionally, the licensee found that the turbine trip bypass was set greater than 30 percent rated thermal power on all RPS channels. The licensee entered the conditions mandated by TS 3.3.1 and calibrated the instruments within the completion time of the conditions. The licensee subsequently issued LER 50-269/2000-05-00 to report this problem.

The inspectors evaluated the RPS channels not being within TS required values using the SDP for Reactor Inspection Findings for At-Power Situations. Because the setpoints only slightly exceeded the TS allowable values, the inspectors determined that this issue was of very low safety significance (Green).

TS 3.3.1 requires three RPS channels to be operable for each function listed in Table 3.3.1-1 which includes the anticipatory trips from the main turbine and main feedwater. Because the RPS channel setpoints for the main turbine and main feedwater anticipatory trips exceeded allowable values for longer than the completion times specified in TS 3.3.1, the licensee was not in compliance with TS LCO 3.3.1. The inspectors considered this a violation of TS 3.3.1. This issue is being treated as an NCV, consistent with Section VI.A.1 of the enforcement policy and is identified as NCV 50-269,287/00-07-03, Reactor Protection System Trip Setpoints Outside Allowable Limits. This violation is in the licensee’s corrective action program as PIP O-00-03607.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspectors reviewed radiological surveys, access controls and verified their implementation for EOC Refueling Outage work for Unit 1 (U1EOC-19). The work was conducted in accordance with Radiation Work Permits. Selected Problem Evaluation Reports (PIP’s) Nos: 0-00-04190, 0-00-04209, 0-00-04218, 0-00-04220, 0-00-04226, 0-00-04200, 0-00-04212, were reviewed for assignment of responsibility, evaluation and timely closure.

Pre-job briefings, work-in-progress, and Health Physics (HP) technician job coverage were observed. The inspectors reviewed procedure HP/0/B/1000/093, Non-Routine Surveillance Requirements, Rev. 10, and observed the planning and control for the high risk dose potential evolutions for the incore probe movements to park, incore cutting, and fuel movement through the transfer canal. Personnel dosimetry results and exposure investigation reports were reviewed and discussed in detail. Licensee activities were reviewed against Updated Final Safety Analysis Report (UFSAR), TS, and 10 CFR Part 20 requirements.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspectors reviewed the plant Refueling Outage Unit 3 EOC-18 (U3EOC-18) ALARA Report, including shutdown chemistry crud burst procedure CP/0/B/2002/010, Addition of Hydrogen Peroxide to the Reactor Coolant System, Rev. 21, and clean-up results. The inspectors reviewed outage job ALARA WORK PLAN dose estimates for the U1EOC-19 Refueling Outage, results of ALARA efforts during the current outage, and dose controls used to track and minimize worker doses for the following jobs: reactor coolant pump seal change-out; steam generator eddy current; installation of nozzle dams; steam generator manway removal; and erection of scaffolds. Shutdown dose rates at selected locations for the past four Unit 1 EOC refueling outages were reviewed. ALARA emergent work planning, work controls and worker dose estimates for the thermocouple leak repair on the reactor head were reviewed including ALARA Radiation Protection Policy Manual ALARA Policy III-04, Rev.0 and System ALARA Manual, Section IV ALARA Planning, Rev. 11. The inspectors also attended daily outage planning meetings.

The inspectors toured the auxiliary, turbine, reactor containment, and radwaste building and the inspectors independently verified dose rates, area surveys, and postings at selected locations. Licensee activities were reviewed against UFSAR, TS, SLC, and 10 CFR Part 20 requirements.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Shipping

a. Inspection Scope

The inspectors reviewed the licensee's facilities, processes and programs for the collection, processing, treatment, shipping, storage and disposal of radioactive materials and radwaste. The inspectors conducted reviews of the following: in-plant liquid and solid waste systems: waste processing and sampling program; shipment activities and records; assurance of quality, including corrective action reports; and training.

Systems reviews, included system descriptions in Chapter 11 of the FSAR, control panel review, facilities tours, liquid waste and recycle system flow diagrams and a review of system changes in accordance with 10 CFR 50.59. The inspectors also toured abandoned in-place radwaste equipment and facilities, and interim storage locations use for processed radwaste.

The inspectors reviewed the licensee's Process Control Program (PCP), including: process documentation; scaling factors (derivation, sampling type, sampling frequency, and effect of changing plant conditions); and determination of waste characteristics and waste classification. The following procedures were reviewed: HP/O/B/1000/089, Resin Sluice Surveillance, Rev. 08; and SH/O/B/2004/001, Preparation and Shipment of Radioactive Material, Rev. 01 and 02.

The inspectors selected five solid radwaste shipping records for detailed review against the requirements contained in 10 CFR Parts 20, 61 and 71, and 49 CFR Parts 100-177. The inspectors reviewed the Oconee Nuclear Station 10CFR61 Manual, Principal Supporting Documentation, Rev. 07. The shipments selected included processed resins and dry active waste shipments. The shipments were Uniform Low-Level Radioactive Waste Manifest Shipment Nos. ONS 00-2036, ONS 00-2023, ONS 00-2026 and ONS 00-2034. The inspectors observed the package preparation, radiation surveys, shipping record preparation and driver instructions for shipment ONS 00-2055.

The inspectors reviewed the licensee's program for assurance of quality in the radwaste processing and radioactive materials transportation program by reviewing, quality assurance audit (99RPS2R2), quality surveillances, and ten PIPs involving the radwaste and transportation program (G-99-00174, G-00-00123, 0-00-01067, 0-00-02114, 0-00-00826, 0-00-03138, 0-00-03644, 0-00-03098, 0-00-02229, 0-00-02821.)

The inspectors reviewed the licensee's program for training personnel involved in the radwaste and radioactive materials transportation program with regard to the requirements contained in NRC IE Bulletin 79-19 and DOT 49 CFR, Subpart H. The inspectors attended procedure revision training for shipping procedures and reviewed the training test questions.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA3 Event Follow-up

(Closed) Licensee Event Report (LER) 50-269/00-005-00: Reactor Protective System Setpoints Calibrated Outside Technical Specification Limits

This event is discussed in Section 1R22.2 of this report. Several as-left setpoints for the RPS anticipatory trips from the main turbine and main feedwater systems were calibrated outside of the allowable values in TS Table 3.3.1-1. The licensee attributed the root cause of this event to an ineffective procedure review during the conversion process from custom to improved TS. Based on the significance review and corrective action program references in Section 1R22.2 of this report, this LER is closed.

4OA6 Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. M. Nazar, Engineering Manager, and other members of licensee management at the conclusion of the inspection on January 9, 2001. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Regulatory Conference Summary

On September 7, 2000, a regulatory conference (SDP/EA-00-137) was held in the Region II Office with the licensee to discuss, in part, an apparent violation (EEI) related to a calculation for the design of a suction source for the high pressure injection system (EEI 50-269,270,287/00-11-01). The NRC concluded that the issue described in the apparent violation represented a violation of NRC regulations. Duke was notified of the NRC's final determination in a letter dated November 10, 2000, in which, a Notice of Violation (NOV) was issued. Accordingly, the EEI is closed and for tracking purposes the violation is identified as VIO 50-269,270,287/00-07-04: Hydraulic requirements had not been adequately considered as design inputs for calculation OSC 3873, Hydraulic Model of High Pressure Injection System with Suction from the Fuel Pool.

PARTIAL LIST OF PERSONS CONTACTED

Licensee
T. Coutu, Superintendent of Operations
T. Curtis, Mechanical System/Equipment Engineering Manager
J. Forbes, Station Manager
W. Foster, Safety Assurance Manager
D. Hubbard, Modifications Manager
C. Little, Civil, Electrical& Nuclear Systems Engineering Manager
W. McCollum Site Vice President, Oconee Nuclear Station
B. Medlin, Superintendent of Maintenance
M. Nazar, Manager of Engineering
L. Nicholson, Regulatory Compliance Manager
M. Thorne, Emergency Preparedness Manager
J. Twiggs, Manager, Radiation Protection
J. Weast, Regulatory Compliance

NRC

D. LaBarge, Project Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-269,270,287/00-07-04	VIO	Hydraulic requirements had not been adequately considered as design inputs for calculation OSC 3873, Hydraulic Model of High Pressure Injection System with Suction from the Fuel Pool (Section 4OA6.2)

Opened and Closed During this Inspection

50-269,270,287/00-07-01	NCV	Inadequate Corrective Actions on BWST Level Instrument Heat Trace (Section 1R01)
50-269/00-07-02	NCV	Failure to Document the Results of a Failed Test and Failure to Document a Test Failure (Section 1R19.2)
50-269,287/00-07-03	NCV	Reactor Protection System Trip Setpoints Outside Allowable Limits (Section 1R22.2)

Previous Items Closed

50-269/00-005-00	LER	Reactor Protective System Setpoints Calibrated Outside Technical Specification Limits (Section 4OA3)

0-269,270,287/00-11-01	EEI	Hydraulic requirements had not been adequately considered as design inputs for calculation OSC 3873, Hydraulic Model of High Pressure Injection System with Suction from the Fuel Pool (Section 4OA6.2)

<u>Discussed</u>

50-269/00-05-11	URI	Operation in Mid-Loop with Containment Purge valves that Subsequently failed to Hold Design Pressure (second example of similar issue) (Section 1R20.2)

LIST OF ACRONYMS USED

AC	-	Alternating Current
ALARA	-	As Low As Reasonably Achievable
ASME	-	American Society of Mechanical Engineers
BAMT	-	Boric Acid Mix Tank
BS	-	Building Spray
BWST	-	Borated Water Storage Tank
CBAST	-	Concentrated Boric Acid Storage Tank
CC	-	Component Cooling
CFR	-	Code of Federal Regulations
CT	-	Current Transformer
DBD	-	Design Basis Document
DC	-	Direct Current
ECCS	-	Emergency Core Cooling System
EOC	-	End of Cycle
F	-	Fahrenheit
FSAR	-	Final Safety Analysis Report
GL	-	Generic Letter
gpm		Gallons per Minute
HPI	-	High Pressure Injection
IP	-	Inspection Procedure
KHU	-	Keowee Hydro Unit
KV	-	Kilovolt
LAR	-	License Amendment Request
LCO	-	Limiting Conditions for Operation
LER	-	Licensee Event Report
LLRT	-	Local Leak Rate Test
LOCA	-	Loss Of Cooling Accident
LPI	-	Low Pressure Injection
LPSW	-	Low Pressure Service Water
LTOP	-	Low Temperature Overpressure Protection
MCC	-	Motor Control Center
NCV	-	Non-Cited Violation
NRC	-	Nuclear Regulatory Commission

NRR	-	Nuclear Reactor Regulation
NSD	-	Nuclear System Directive
PIP	-	Problem Investigation Process
PMT	-	Post-Maintenance Testing
PRA	-	Probabilistic Risk Assessment
psig	-	pounds per square inch gauge
QA	-	Quality Assurance
QC	-	Quality Control
RBS	-	Reactor Building Spray
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RCW	-	Raw Cooling Water
RPS	-	Reactor Protection System
SDP	-	Significance Determination Process
SLC	-	Selected Licensee Commitments
SR	-	Surveillance Requirement
SRA	-	Senior Risk Analyst
SSC	-	Structure, System and Component
SSF	-	Standby Shutdown Facility
TDEFW	-	Turbine Driven Emergency Feedwater
TS	-	Technical Specification
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item

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

- 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.

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