December 18, 2000

Mr. Theodore Sullivan
Vice President - Operations
Entergy Nuclear Northeast
James A. FitzPatrick Nuclear Power Plant
Post Office Box 41
Lycoming, NY 13093

SUBJECT: NRC'S FITZPATRICK REPORT 05000333/2000-009

Dear Mr. Sullivan:

On November 18, 2000, the NRC completed an inspection at the James A. FitzPatrick Nuclear Power Plant. The results of this inspection were discussed on December 1, 2000, with you and other members of your staff. The enclosed report presents the results of that inspection.

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.

The NRC identified one finding that was evaluated under the risk significance determination process and were determined to be of very low safety significance (Green). This finding 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.

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/ADAMS/index.html (the Public Electronic Reading Room). Should you have any questions regarding this report, please contact me at 610-337-5211.

Sincerely,

/RA/

Glenn W. Meyer, Chief
Projects Branch 3
Division of Reactor Projects

Docket No. 05000333
License No.: DPR-59

Enclosure: Inspection Report 05000333/2000-009

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

REGION I

Docket No.:	05000333
License No.:	DPR-59
Report No.:	2000-009
Licensee:	Entergy Nuclear Northeast Post Office Box 41 Lycoming, NY 13093 (During the inspection period the licensee was the New York Power Authority. Ownership for the facility and the operating licensee transferred on November 21, 2000.)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Post Office Box 41 Scriba, New York 13093
Dates:	October 1, 2000 to November 18, 2000
Inspectors:	R. A. Rasmussen, Senior Resident Inspector R. A. Skokowski, Resident Inspector T. A. Moslak, Health Physicist T. F. Burns, Reactor Inspector P. R. Frechette, Physical Security Inspector
Approved by:	G. W. Meyer, Chief Projects Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000333/2000-009, on 10/01-11/18/2000; New York Power Authority, James A. FitzPatrick Nuclear Power Plant. Operability Evaluations.

The report covers a seven-week inspection by resident inspectors, and inspections of occupational radiation safety, safeguards, and inservice inspections by regional inspectors. These inspections identified one green issue. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP) in Inspection Manual Chapter 0609 (see Attachment 1).

Mitigating Systems

Green: The inspector determined the deficiency and event report (DER) response written to evaluate deficiencies on the high pressure coolant injection (HPCI) system inadequate, because two of the deficiencies that could have had a significant impact on HPCI operability were not adequately addressed. Specifically, the failure of reversing chamber bolts on the interior of the turbine casing was mis-characterized as normal wear. Additionally, damage to the governor speed sensor was attributed to installation damage without an appropriate basis.

The evaluations of these deficiencies were of concern because if not adequately corrected, the conditions could have resulted in HPCI inoperability. However, this inspection finding was considered to have very low safety significance, because after reevaluation the original conclusions were supported and the conditions did not impact HPCI operability. No violation of requirements was identified. (Section 1R15).

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<u>Report Details</u>

SUMMARY OF PLANT STATUS

The plant began the period at full power and operated until the planned shutdown for refueling outage 14 on October 7, 2000. The refueling outage was completed and the reactor restarted on November 12. The plant returned to 100% power on November 17, 2000.

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

1R04 <u>Equipment Alignments</u>

a. <u>Inspection Scope</u>

The inspectors performed the following partial system walkdowns:

- Electrical bus 10500 (safety-related) while bus 10600 was unavailable for planned maintenance.
- Shutdown cooling alignment in the control room following the transition to shutdown cooling.
- Electrical lineup with B and D emergency diesel generators out of service for outage maintenance.

b. <u>Findings</u>

No findings of significance were identified.

1R05 <u>Fire Protection</u>

a. <u>Inspection Scope</u>

The inspectors focused on fire protection equipment during tours of the drywell. In addition, the inspectors reviewed locations where welding or grinding was in progress or where fire barriers had been breached to support maintenance activities.

The inspectors also reviewed the evaluation and corrective actions taken for Deficiency and Event Report (DER) 00-5323, related to a small fire on the east diesel fire pump.

b. <u>Findings</u>

No findings of significance were identified.

1R08 In-service Inspection (ISI)

a. Inspection Scope

The inspector selected samples of nondestructive examination (NDE) and American Society of Mechanical Engineers (ASME), Section XI, code replacement activities for evaluation based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in a significant increase in risk of core damage. Also, the inspector evaluated the corrective action of problems identified during ISI activities. The inspector reviewed a sample of deviation reports from ISI examinations.

The inspector reviewed three types of NDE activities, including volumetric, surface and visual examinations, to verify the effectiveness in monitoring degradation of risk significant systems, structures and components. In addition, the inspector evaluated the disposition of nonconforming conditions identified in the inspection sample and verified analyses were performed for acceptance and continued operation without repair. The inspector performed a review of the ultrasonic (UT) test reports for reactor pressure vessel (RPV) welds VC-TH-2-F, VC-TH-1-2, high pressure core injection (HPCI) weld 10-23-707, magnetic particle (MT) test report for residual heat removal (RHR) field weld three (FW-3, 10SV-74B valve replacement), and the visual examination data sheets and indication notification report of RHR seismic pipe restraint PFSK-2412 for compliance with the requirements of the ASME boiler and pressure vessel code. Also, the inspector reviewed a sample of video recordings of the remote in vessel visual inspection (IVVI) of core spray piping and various welds in the core shroud within the reactor vessel. The inspector confirmed the examination and evaluation of previously identified indications being monitored in the ISI program. The inspector reviewed a sample of indication notification reports from 1998 the visual inspection of the containment liner (coating failure, corrosion and other damage) for compliance with the requirements of ASME Section XI, IWE (requirements for class MC and metallic liners of class CC components).

The inspector reviewed welding activities associated with the replacement of selected components to verify the activities were performed in accordance with the requirements of ASME Section XI and IX. The inspector reviewed the weld procedures (WPS CS-1/1-B and WPS CS-1/1-C) used, procedure and personnel qualification records (PQRs 599, 600, 602, 604 and 604A), and interviewed the welding engineer and other personnel involved in welding activities. The inspector interviewed the licensee’s Level III radiographer and supplemental Level III personnel who performed the interpretation of the radiographs evaluated. Radiographs of welding activities were reviewed for welds FW1 and FW2 for the replacement of feedwater isolation valve 34-NRV-111B, FW1 and FW2 for the replacement of RCIC turbine isolation valve 13-MOV-16, and welds 6-12-912 and 6-12-912A for replacement of the reactor water cleanup valve 12-MOV-18. The review verified flaws were appropriately identified and sized, and their location within the weld(s) were appropriately recorded and evaluated for compliance with the requirements of ASME Section XI.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the implementation of the Maintenance Rule (MR) as related to Licensee Event Report (LER) 99-010, Main Turbine Trip and Reactor Scram Due to Degraded Cable in Main Generator Anti-Motoring Circuit.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work

a. Inspection Scope

The inspectors reviewed the maintenance risk assessments and maintenance activities associated with the following emergent work activities:

- Replacement of the Division B electric plant loss-of-offsite-power/loss-of-coolant-accident test switch following failure of the test switch during surveillance testing.

- Replacement of the HPCI inboard isolation valve disk following test failure.

The inspectors also reviewed the plant conditions and system alignments established for reactor disassembly and refueling operations.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability determinations associated with the following plant equipment challenges:

- Failure of the supply breaker to the 10102 electric bus.
- HPCI operability evaluation, DER 00-04946, which collected and evaluated twelve various deficiencies identified during outage maintenance on HPCI. (The DER summarized the individual deficiencies and evaluated the cumulative effect on operability.)

b. Findings

The inspector considered the DER response written to evaluate deficiencies on the HPCI system inadequate, because two of the deficiencies that could have had a significant impact on HPCI operability were not adequately addressed. Specifically, the failure of reversing chamber bolts on the interior of the turbine casing was mis-characterized as normal wear. Additionally, damage to the governor speed sensor was attributed to installation damage without an appropriate basis. The evaluations of these deficiencies were of concern because if not adequately corrected, the conditions could have resulted in HPCI inoperability. However, this inspection finding was considered to have very low safety significance, because after reevaluation the original conclusions were supported and the conditions did not impact HPCI operability. There were no findings identified for the issue of the 10102 supply breaker.

Two reversing chamber bolts and one tab washer were found missing and others were found loose during the overhaul of the HPCI turbine. The reversing chambers are bolted to the inside of turbine case and are subjected to steam impingement. Loose hardware on the inside of the turbine was a concern due to the potential for damage to other components and the potential for a reduction in performance if the reversing chambers were dislodged. The issue of missing bolts was an industry identified issue that NYPA had taken previous corrective actions for; however, based on the missing bolts these previous efforts appeared to be inadequate. These fasteners were installed during the last HPCI overhaul six years ago.

The DER evaluation by NYPA stated that the missing hardware was considered normal wear and not a failure of any process. The inspectors disagreed with this conclusion and requested that NYPA reevaluate their position. NYPA reevaluated and concluded that the bolting should not fail between overhaul cycles. They evaluated the failure mechanism and concluded that the tabs of the tab lock washer were installed such that steam impingement caused the tabs to bend open, thus allowing the bolts to loosen. Corrective actions recommended by the vendor were to install the lock tabs firmly to the side of the bolts to minimize the area for steam impingement. Additionally, NYPA is planning to inspect the tabs during the next refueling outage. NYPA reevaluated the as found condition and still maintained that HPCI would have functioned if called upon. This was now based on the facts that the fasteners were expelled without significant damage to the turbine and the reversing chambers were adequately restrained by the remaining fasteners. The inspectors considered this evaluation adequate.

The second issue of damage to the speed pickup sensor was characterized as an installation error without an adequate evaluation of the failure. The speed sensor is a magnetic sensor that generates electronic pulses as gear teeth pass by the sensor. In this application the gap between the sensor and gear is very small (0.008"). The observed damage indicated that the sensor had struck the gear, thus causing the gear to slip on the shaft and damage to the teeth and sensor. However, as the teeth wore down and provided clearance, the sensor continued to function. Although this condition could have been caused by an installation error, the inspector considered it significant to eliminate the possibility that the damage occurred in service. The inspector was concerned that the damage could have occurred with the turbine in service due to wear in the gear shaft bushings that could have allowed the gear to contact the teeth. If the damage had occurred with the unit in service due to play in the gear shaft, the condition could have resulted in the failure of HPCI. Subsequent to the inspector's questions, NYPA reviewed maintenance records and interviewed the mechanics that performed the overhaul. NYPA concluded that the shaft bushings were within specification and the gear could not have hit the sensor unless it was installed improperly. The inspector considered the evaluation adequate.

The inspector considered the evaluation of DER 00-04946 inadequate in that the root cause of the missing bolting and the damaged speed sensor did not assure the appropriate bases for conclusions were used. However, these issues were considered to have very low safety significance because after reevaluation, the original conclusions were supported and the conditions did not impact HPCI operability. Although considered an inspection finding, this issue was not a violation of NRC requirements.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modifications that were installed during the refueling outage:

- JD-99-020, EPIC Rods In Monitor.
- JD-99-089, Main Steam Isolation Valve Enhancements.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post maintenance tests:

- ST-24Q, RCIC [reactor core isolation cooling] Turbine Slow Roll and Overspeed Test.
- TOP-85, Flushing of Reactor Vessel Bottom Drain.
- ST-39H, RPV System Leakage Test and CRD Class 2 Piping InserviceTest (ISI).

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors reviewed and/or observed various risk significant activities associated with the refueling outage. These inspections included:

- Reviewed the pre-outage risk assessment of planned work.
- Observed portions of the reactor shutdown and cooldown.
- Reviewed the availability and technical adequacy of reactor water level and temperature indicating instrumentation.
- Reviewed the availability of protected equipment specified by shutdown risk assessment.
- Reviewed contingency plans as specified by NYPA's shutdown risk assessment.
- Verified tagged out equipment was in the correct position as described by the associated tag.
- Toured spaces normally inaccessible during power operation.
- Observed refueling operations from the refuel bridge, and reviewed the completed refueling checklist.
- Observed portions of the core verification following fuel reload. In addition, based on discussions with reactor engineers, the inspectors selected six fuel assemblies with a high risk to fuel barrier damage if they were incorrectly located in the core, and the inspector independently verified proper location and orientation.
- Observed portions of ST-39H, the reactor pressure vessel system leakage test.
- Observed portions of the reactor startup.

In addition, the inspectors reviewed NYPA's evaluations and corrective actions for the following:

- DER 00-04968, IVVI Fuel Move Resulted in the Potential for a Control Rod to Lean.
- DER 00-05434, Mis-orientation of Fuel Bundles Identified during Core Verification.
- DER 00-05467, Repairs to 23MOV14, following local leak-rate test (LLRT) failure.
- DER 00-05055, Evaluation of Hardening Grease in motor-operated valves (MOVs).
- DER 00-04878, Pipe Support Discrepancies
- DER 00-04496, ISI Inspection Deficiencies
- DER 00-05312, Core Shroud ISI Deficiencies

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed procedures and observed portions of testing related to the following surveillance tests:

- ST-24J, RCIC Flow Rate and Inservice Test (IST).
- ST-9C, Emergency AC Power Load Sequencing Test and 4kV emergency Power System Voltage Relays Instrument Functional Test.
- ST39B, Containment Isolation Valve Local Leak Rate Testing.
- ST-2HA, LPCI [low pressure coolant injection] Initiation Logic System Functional Test.

The inspectors also reviewed NYPA's response to ACTS item 00-53969, regarding the adequacy of the acceptance criteria for the 125 Vdc battery service test.

b. Findings

No findings of significance were identified.

1R23 Temporary Modifications

a. Inspection Scope

The inspectors reviewed documentation associated with temporary modification 00-069, which allowed operation of drywell continuous atmosphere monitor (DW CAM) system with one particulate sample pump per division as corrective actions to reduce flow oscillations.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls

a. Inspection Scope

During the period October 30 through November 3, 2000, the inspector conducted the following activities to determine the effectiveness of administrative, operational, and engineering controls to minimize and equalize personnel exposure for tasks during the refueling outage.

The inspector reviewed pertinent information regarding cumulative exposure history, current exposure trends, and ongoing activities in order to assess the effectiveness in establishing exposure goals, and in keeping actual exposure as low as is reasonably achievable (ALARA).

The inspector reviewed the associated exposure controls specified in ALARA Reviews (AR) for selected jobs. The actual cumulative exposure was compared with the estimated exposure and evaluated using the criteria contained in the relevant NRC's Significance Determination Process. Jobs that were reviewed included defuel/refuel/IVVI/LPRM activities (AR00-031), control rod drive replacement (AR00-033), and traversing incore probe system upgrades (AR00-054).

Independent radiation surveys were performed in areas of the drywell, reactor building, and waste processing building to confirm posted survey results and assess the adequacy of radiation work permits and associated controls. Keys to technical specification locked high radiation areas were inventoried, and these areas were verified to be properly secured and posted during plant tours.

Individual exposure records were reviewed for completed tasks and for those currently in progress. Included in this review were the exposure records for workers who performed control rod drive (CRD) change outs, a declared pregnant worker, maintenance personnel, and radiation protection technicians. Interviews were conducted with a mechanical maintenance supervisor and a health physics supervisor to assess departmental efforts to minimize and equalize dose to their respective staffs.

The inspector attended daily radiation protection department staff meetings and site outage planning meetings. On October 31, 2000, the inspector observed pre-job RWP briefings for reactor cavity decontamination and for replacement of the drywell dome O rings.

The effectiveness of various management controls were evaluated by reviewing the Quality Assurance Audit No. A00-13J, regarding radiological controls implemented during the outage, and the results of management observations for various outage tasks, including processing a highly activated CRD for disposal.

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

- DER 00-4947, Worker entered a locked high radiation area without wearing a remote dosimeter.
- DER 00-5170, Failure to notify the control room before removing CRDs.
- DER 00-5273, LPRM hot end near water surface.
- DER 00-5298, Poor ALARA practices by uncoupling CRDs from under the vessel instead of from the refuel floor.
- DER 00-5310, Outage dose greater than outage goal.
- DER 00-5399, Failure to maintain CRD changeout project ALARA.
- DER 00-5475, Dose rate increase identified on fuel pool cooling and decay heat removal system components.
- DER 00-5455, Improvements for transferring an irradiated CRD.
- DER 00-5461 & 4772, Contamination of reactor building floors.

- DER 00-5479, Worker working in the Radiological Control Area without electronic dosimetry.
- DER 00-5489, ALARA goal has been exceeded for the outage and for the year.
- DER 00-5564, Dose accrued for CRD changeout exceeded estimate.
- DER 00-5599, Cumulative exposure is approaching NRC significance determination process decision point.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS
Cornerstone: Physical Protection

3PP1 Access Authorization Program

a. Inspection Scope

The following activities were conducted to determine the effectiveness of the licensee's behavior observation portion of the personnel screening and fitness-for-duty programs:

Five supervisors representing the maintenance, operations, radiation protection, system engineering and electrical departments were interviewed on October 4, 2000, regarding their understanding of behavior observation responsibilities and the ability to recognize aberrant behavior traits. The inspector reviewed two (2) Access authorization/ fitness-for-duty self-assessments, an audit, event reports, and loggable events for the four previous quarters. On October 4, 2000, five (5) individuals who perform escort duties were interviewed to establish their knowledge level of those duties. Behavior observation training procedures and records were also reviewed.

b. Findings

No findings of significance were identified.

3PP2 Access Control

a. Inspection Scope

The following activities were conducted during the period October 2-6, 2000 to verify that NYPA had effective site access controls and equipment in place designed to detect and prevent the introduction of contraband (firearms, explosives, incendiary devices) into the protected area:

A random sample of personnel granted unescorted access to the protected and vital areas were checked to assure that they were properly screened, identified and authorized. Program enhancements in the area of access authorization instituted as a result of the pilot program were also reviewed. Site access control activities were observed, including personnel and package processing through the search equipment at the access point during peak ingress periods on October 3 and 4, 2000, and vehicle searches, on October 5, 2000. On October 3, 2000, testing of all access control equipment, including the metal detector, explosive material detectors, and X-ray examination equipment, was observed. The Access Control event log, an audit, and three (3) maintenance work requests were also reviewed.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspector reviewed NYPA's programs for gathering and submitting data for the fitness-for-duty, personnel screening, and protected area security equipment performance indicators. The review included the tracking and trending reports, personnel interviews and security event reports for the performance indicator data submitted from the 2nd quarter of 1997 through the 2nd quarter of 2000.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

The inspection finding in section 1R15, the inadequate response to the HPCI DER, also had implications regarding problem identification, evaluation and resolution. Additional items associated with NYPA's corrective action program were reviewed without findings.

4OA5 Other

.1 Review of INPO report: The inspectors reviewed the Institute of Nuclear Power Operators (INPO) report for the evaluation conducted May 1-12, 2000. The final report was issued on June 23, 2000. The findings were consistent with NRC findings and no new issues were identified.

.2 (Closed) LER 50-333/00-014: Failure to Meet Standby Gas Treatment Technical Specifications Required Action Statement Test Interval Requirement. This personnel error was of a short duration and had no impact on safety. This LER pertained to a minor finding and was closed during an in-office review.

.3 (Closed) LER 50-333/00-004-03: RCIC System Inoperable for Greater than Seven Days and Inoperable During Two Plant Start-Up Evolutions. This LER revision was submitted to report the results of the equipment failure evaluation and corrective actions. No new issues were identified and this LER was closed during an in-office review.

4OA6 Meetings

Exit Meeting Summary

On December 1, 2000, the inspectors presented the inspection results to Mr. Sullivan and other members of his staff, who acknowledged the finding presented and did not contest any of the inspectors' conclusions. Additionally, the inspectors confirmed that none of the information reviewed by the inspectors was considered proprietary.

PARTIAL LIST OF PERSONS CONTACTED

M. Anderson	Supervisor, Health Physics Instrumentation/Respiratory Protection
T. Bergene	Supervisor, Radiation Protection Operations
R. Brown	Radiation Protection Supervisor
G. Brownell	Licensing Engineer
L. Burrows	Radiation Protection, Technician
M. Colomb	Site Executive Officer
J. Haley	Security Manager
K. Hobbs`	Supervisor, Dosimetry
D. Kieper	General Manager Maintenance
J. LaPlante	Security Supervisor
D. Lindsay	Plant Manager
C. Moreau	Quality Assurance Assessor
R. Murray	Radiation Protection, Technician
W. O'Malley	General Manager Operations
K. Pushee	Radiation Protection Manager
W. Rohr	ALARA Engineer
D. Ruddy	Manager Design Engineering
T. Sullivan	VP Operations
G. Tasick	Licensing Manager
G. Thomas	Director Design Engineering
A. Zaremba	General Manager Support Services

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>
None

<u>Opened and Closed</u>
none

<u>Closed</u>

LER 50-333/00-014: Failure to Meet Standby Gas Treatment Technical Specifications Required Action Statement Test Interval Requirements.

LER 50-333/004-03: RCIC System Inoperable for Greater than Seven Days and Inoperable During Two Plant Start-Up Evolutions.

LIST OF ACRONYMS USED

ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
CAM	Continuous Atmosphere Monitor
CFR	Code of Federal Regulations
CRD	Control Rod Drive
DER	Deficiency and Event Report
DW	Drywell
FSAR	Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
IR	Inspection Report
ISI	Inservice Inspection
IST	Inservice Test
IVVI	In-Vessel Visual Inspection
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
LLRT	Local Leak-rate Test
MOV	Motor-operated Valve
MR	Maintenance Rule
MT	Magnetic Particle Test
NDE	Non-destructive Examination
NRC	Nuclear Regulatory Commission
NYPA	New York Power Authority
PI	Performance Indicator
PIM	Plant Issues Matrix
RHR	Residual Heat Removal
RCIC	Reactor Core Isolation Cooling
RPV	Reactor Pressure Vessel
SDP	Significance Determination Process
UT	Ultrasonic Test

APPENDIX 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

- 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 margins.

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.

ATTACHMENT 1, Documents Reviewed

Documentation Reviewed for Inspection Procedure 71111.08, In-service Inspection Activities

NDT Examination Reports

R014-UT-031	Ultrasonic test of weld VC-TH-2-F, RPV head to flange
R014-UT-024	Ultrasonic test of weld 10-23-707, HPCI pipe to elbow
R014-UT-028	Ultrasonic test of weld VC-TH-1-2, RPV head to dome
JAF-RPT-PC-03089	IWE Inspection Summary Report
Magnetic Particle	00S093 10SV-74B RHR Safety Relief Valve Replacement-FW3

Radiograph Review

Radiographic	98R058	FW Isolation Valve Replacement - FW1
Radiographic	98R059	FW Isolation Valve Replacement - FW-2
Radiographic	98R060	RCIC Valve Replacement - Weld 89-202 FW1
Radiographic	98R061	RCIC Valve Replacement - Weld 89-202 FW2
Radiographic	00R024	RWC MOV Valve Replacement - 6-12-912
Radiographic	00R025	RWC MOV Valve Replacement - 6-12-912A

Deviation/Event Report

DER 00-05044	N9-C1 Nozzle (CRD) to Cap Weld Unacceptable (Cracking)
DER 00-05134	IGSCC crack like indication on B core spray internal piping
DER 00-05312	IVVI core shroud exam - several crack like indications

Action/Commitment Tracking

ACT 00-53941Action required for disposition of DER 00-05044
ACT 00-53942Effectiveness of Corrective Action of DER 00-05044
ACT 00-54053Approval of disposition (repair) of flaw identified be DER 00-05044
ACT 00-54211Reexamination of weld #12 crack-like indication from RO14
ACT 98-37745Examine indication from RO13 in core spray piping
ACT 98-37823Examine crack like indications in core shroud welds from RO13
ACT 00-54034Response to DER 00-05134 Analysis of crack like indication core spray

Weld Procedure Specification and Procedure Qualification Record

WPS CS-1/1-B	P1 to P1 Groove and Fillet Welding GTAW
WPS CS-1/1-C	P1 to P1 Groove and Fillet Welding SMAW
PQR	599,600,602,604,604A Supporting CS1/1B and C

Indication Notification Report

R014-INR-002 RHR Seismic Constraint numerous recordable indications
R014-INR-006 Control Rod Drive Cut & Cap (N-9-C1)
JAF-IWE-98-014 Liner paint flaking
JAF-IWE-98-019 Liner Gouge
JAF-IWE-98-021 Arc strikes on liner
JAF-IWE-98-101 Liner paint blisters

Repair Work Package

WP 00-01660-01 Repair of wall thinning - RHR strainer housing

Calculation

JAF-CALC-RHR-04045 Minimum wall thickness calculation-RHR Strainer housing
JAF-CALC-RHR-04178 RHR pipe support - load carrying capability evaluation

Documents reviewed for procedure 71130, Access Authorization Program and Access Control Program

J. A. FitzPatrick Security Program Audit, A00-06J, June 26, 2000
J. A. FitzPatrick Fitness for Duty Program Audit, A00-04J, May 8, 2000
Fitness for Duty Lesson Plan. (GET training program)
Continuous Behavior Observation Program Guide
Security Event Log
Semi-Annual Fitness for Duty reports