January 30, 2001

Mr. A. Alan Blind
Vice President - Nuclear Power
Consolidated Edison Company of
New York, Inc.
Indian Point 2 Station
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: NRC's INDIAN POINT 2 INSPECTION REPORT 05000247/2000-014

Dear Mr. Blind:

On December 30, 2000, the NRC completed an inspection at the Indian Point 2 reactor facility. The enclosed report presents the results of that inspection. This inspection report covers a variety of issues during the period approaching the restart of Indian Point 2. The results of this inspection were discussed on January 25, 2001, with Mr. A. Blind and members of your staff.

NRC inspectors examined numerous activities as they related to reactor safety and compliance with the Commission's rules and regulations, and with the conditions of your operating license. The inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel. Specifically, it involved seven weeks of resident and region-based inspections of engineering, operations and maintenance, radiation protection, security, and weld radiographs associated with the steam generator replacement project.

During this inspection period, a team of inspectors completed an inspection of design controls and vendor interface problems at Indian Point 2. The inspection took place December 11-15 and 19-22, 2000 using procedure 71111.21(OA), Safety System Design and Performance Capability. The NRC team identified three issues, two of which were evaluated under the risk significance determination process (SDP) and were determined to be of very low safety significance (green). The issues were determined to involve violations of NRC requirements in the areas of design control and corrective action. 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.1.A of the NRC's Enforcement Policy. Although these issues are of very low safety significance, as stated in an NRC status letter to you, dated December 22, 2000, these weaknesses in your design control process continue, and provide further examples of weaknesses in your corrective action program.

Additionally, also included in this report is a separate issue identified during NRC review of the safety injection system. This issue involved corrective actions that were ineffective to prevent recurrence of material condition concerns which was evaluated under the SDP to be of very low safety significance (green). For similar reasons as discussed above, this issue is being treated as a non-cited violation. If you deny the non-cited violations included in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Indian Point Unit 2 facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter 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 Mr. Peter Eselgroth at 610-337-5234.

Sincerely,

/RA/

Brian E. Holian, Deputy Director
Division of Reactor Safety

Docket No. 05000247
License No. DPR-26

Enclosure: Inspection Report 05000247/2000-014

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REGION I

Docket No.:	05000247
License No.:	DPR-26
Report No.:	05000247/2000-014
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	November 19, 2000 to December 30, 2000
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector Todd Fish, Reactor Engineer Steve Dennis, Reactor Engineer Marc Ferdas, Reactor Engineer John Hanna, Resident Inspector Richard Skokowski, Resident Inspector Alan Blamey, Resident Inspector Lois James, Resident Inspector Frank Arner, Senior Reactor Engineer Melvin Holmberg, Senior Reactor Engineer Kenneth Jenison, Reactor Engineer James Trapp, Senior Reactor Analyst George Morris, Reactor Engineer Aniello Della Greca, Senior Reactor Engineer Suresh Chaudhary, Senior Reactor Engineer Michael Modes, Senior Reactor Engineer Paul Frechette, Security Inspector
Approved by:	Peter W. Eselgroth, Chief Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS
Indian Point 2 Nuclear Power Plant
NRC Inspection Report 05000247/2000-014

IR 05000247-00-14, on 11/19 - 12/30/2000; Con Edison; Indian Point 2 Nuclear Power Plant. Resident Operations Report, Physical Protection, Emergency Preparedness; Steam Generator Replacement Project, Control of Licensing Basis, and Plant Restart Readiness.

The inspection was conducted by resident and region-based inspectors. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP). No significant findings or issues were identified in the Reactor Safety Area. This inspection identified one “no color” and three green issues. The “no color” significance level indicates that the IMC 609 “Significance Determination Process” does not apply to this finding.

Cornerstone: Mitigating Systems

Green. The licensee did not have a formal process for implementing changes to the plant licensing basis, and certain limits and provisions of two technical specification amendments were not adequately incorporated into plant operating procedures. As a result, there was the potential to have exceeded the technical specification analytical limits on safety injection accumulator pressure, and post-accident radiological doses to control room operators could have exceeded analyzed limits. The conditions had a potential impact on safety in that fuel peak cladding temperature and control room habitability could have been adversely affected. If left uncorrected, inadequate implementation of license amendments could result in a more significant safety concern. The conditions were evaluated using the NRC’s significance determination process as having very low safety significance because no actual loss of safety function occurred. This violation of the design control requirements of 10 CFR 50, Appendix B, Criterion III was treated as a non-cited violation. (Section 1R21.3)

Green. The licensee does not have formal procedures to control the verification, validation, and supply of input data and assumptions to the NSSS vendor, and administrative controls were not adequate to ensure that accident analysis input assumptions were not invalidated by plant modifications. As a result, discrepancies existed between the values assumed in certain accident analyses and actual plant conditions and procedure limits. The discrepancies had potential impact on post-accident fuel peak cladding temperature and containment peak pressure. If left uncorrected, the lack of formal control of design inputs could become a more significant safety concern. The specific conditions caused by the lack of formal design controls were evaluated using the NRC’s significance determination process as having very low safety significance because of the limited actual consequences of the input discrepancies on the accident analysis conclusions, and no loss of safety function occurred. This violation of the design interface control requirements of 10 CFR 50, Appendix B, Criterion III was treated as a non-cited violation. (Section 1R21.4)

Summary of Findings (cont'd)

Cross-Cutting Issues: Problem Identification and Resolution

Green. Corrective actions were ineffective to prevent recurrence of material condition concerns with the freeze protection for the refueling water storage tank (RWST), primary water storage tank (PWST) and condensate storage tank (CST) level switches. Over the last three years several condition reports associated with the material condition of the freeze protection for these level switches had been generated, some of which were associated with actual failures of the switches. Although in each case corrective actions were taken to address specific failures, no corrective actions were taken to prevent recurrence of problems with the freeze protection of these level instruments. This issue had a very low safety significance because it did not result in the actual loss of a safety function. The failure to take corrective actions to preclude repetition is being treated as a non-cited violation of 10CFR50, Appendix B, Criterion XVI, “Corrective Action.” (Section 1R21.6)

No Color. The NRC identified that the lack of formal design interface controls that are required by Criterion III of 10 CFR 50, Appendix B, and the licensee’s Quality Assurance Program Description had been identified previously by the licensee’s Quality Assurance organization and the NRC. Failure to promptly correct this condition adverse to quality resulted in multiple discrepancies between design inputs used in accident analyses and actual plant conditions and procedures. The matter had a potential impact on safety due to the potential effects on safety margins, which left uncorrected could become a more significant safety concern. This issue had a very low safety significance because the design discrepancies involved did not result in the actual loss of safety function. This violation of the corrective action requirements of 10 CFR 50, Appendix B, Criterion XVI was treated as a non-cited violation consistent with Section VI.1.A of the Enforcement Policy due to the very low safety significance of the specific design discrepancies involved. (Section 40A2)

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Attachment

Report Details

SUMMARY OF PLANT STATUS

At the start of the inspection period, the plant was in a cold shutdown condition to refuel the reactor following the replacement of the four steam generators. After completing restart readiness reviews and preparations, Con Edison heated the reactor coolant system up to normal operating temperature and pressure. The operators brought the reactor critical at 3:50 a.m. on December 30, 2000. Zero power physics testing was in progress at the end of the inspection period.

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

1R04 Equipment Alignments

.1 Partial System Walkdowns

a. Inspection Scope (71111.04)

The inspectors performed partial system walkdowns to verify system and component alignment and note any discrepancies that would impact system operability. The inspection included a review of corrective actions for discrepant conditions. The review was completed for the following:

- Emergency Diesel Generators
- Essential Service Water Equipment Alignment
- Safety Injection System
- Containment Spray
- Weld Channel and Penetration Pressurization System
- Isolation Valve Seal Water System
- Power Range Nuclear Instrumentation

The inspectors performed a partial walkdown of selected essential service water header components to determine if the valves and associated instrumentation were correctly configured. The inspectors verified service water valve positions and instrumentation associated with the emergency diesel generators, instrument air coolers, fan cooler units, and the zebra mussel/ residual chlorine monitors.

The inspectors performed partial walkdowns for the safety injection system, containment spray system, weld channel and penetration pressurization system, and the isolation valve seal water system. The inspectors verified system alignments in accordance with check-off list (COL) 10.0, “Locked Safeguards Valves.”

The inspectors observed operators restore a power range nuclear instrument (N-44A). The power range instrument was being restored following zero power physics testing. Operators used system operating procedure (SOP) 13.1, "Nuclear Instrumentation System Operations."

b. Issues and Findings

There were no findings identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a detailed system walkdown of the auxiliary feedwater (AFW) system to verify equipment alignment. In addition, the review included, but was not limited, to Updated Final Safety Analysis Report sections 10.2.6.3, Auxiliary Feedwater System, and 14.1.9, Loss of Normal Feedwater; open work orders (41); open conditions reports (47); the modification for the replacement of AFW flow control valves FCV-405A, B, C, D, and FCV-406A, B, C, D; and SE-99-091-MM, to identify discrepancies that would impact system operability.

b. Issues and Findings

10 CFR 50.59 safety evaluation SE-99-091-MM associated with the modification to replace the auxiliary feedwater regulating valves did not evaluate a potential for flow blockage due to debris. The safety evaluation did not evaluate the valves' disk stack potential for clogging, thereby reducing AFW flow to the steam generators. The replacement valves have a different size configuration in the stacked disc than the previous globe valves. Con Edison initiated condition report 200010402 to evaluate this condition. Additionally, due to concerns over deterioration of the city water line (ref. Inspection Report 05000247/2000-13), the inspector questioned whether the quality of the back-up AFW supply source was adequately evaluated. In May 2000, Con Edison issued CR 200003581 to address corrosion from the city water supply to the auxiliary feedwater system.

Con Edison revised safety evaluation SE-99-091-MM. The revised safety evaluation concluded that no increased potential existed for debris to block flow. The basis for this conclusion was that the stacked disc openings in the globe valve and the replacement drag valve were similar (0.073 inch vs. 0.074 inch), the minimization of carryover due to two vertical eight foot risers in the suction piping, installation of dual inlet strainers in the new flow control valve, and the installation of inside diameter grooves which function to capture small particles. The revised safety evaluation adequately addressed the inspectors question on the quality of the 10 CFR 50.59 safety evaluation.

1R05 Fire Protection

a. Inspection Scope (71111.05)

The inspectors performed walkdowns of various plant areas to assess Con Edison's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. The areas included:

- Auxiliary Feedwater Pump Room

- Emergency Diesel Generator Building
- Charging Pumps 21, 22, and 23 (Fire Zones 5, 6, and 7,)
- 80 foot primary auxiliary building (Fire Zones 7A and 8)

b. Issues and Findings

There were no significant findings identified. The inspector noted Con Edison had identified errors in the pre-fire plan drawings that describe the location of emergency battery lights in plant fire zones. The errors were not significant since the pre-fire plan drawings are not used for design changes or fire brigade response and are not referenced in alternate safe shutdown procedures. Condition report 200007006 described corrective actions to eliminate location of emergency lights on the pre-fire plan since that information is available in other plant records.

The inspector confirmed licensee actions in response to an unsealed conduit as documented in NRC report 05000247/2000-011. The open penetration was not a fire seal, however it could have been an unmonitored release pathway from the electrical penetration room. Con Edison installed a cap on the open penetration and the inspector confirmed the licensee actions.

1R07 Heat Sink Performance

a. Inspection Scope (71111.07)

Service Water Heat Exchanger Performance

The inspector reviewed the following documents and heat exchanger performance tests to verify that the test acceptance criteria and results, the frequency of testing to detect potential degradation, and testing methodology was sufficient to detect degradation of heat removal capabilities, such as from bio-fouling:

- Technical Report No. 97222-TR-28, Indian Point Unit 2 GL-89-13 Heat Exchanger Performance Assessment Program
- Indian Point Station Generic Letter 89-13 Service Water Program
- PT-2Y10A, 21 CCW HX Test, dated November 5, 1999
- PT-2Y10B, 22 CCW HX Test, dated November 8, 1999
- Calculation No. 00204-C-001, Component Cooling Water Heat Exchanger Performance Evaluation
- PT-2Y9D, 24 FCU HX Test, dated December 18, 1998
- Calculation No. PGI-00397-00, Evaluation of 24 FCU Test PT-2Y9D
- Test Instrument Accuracies (CR 200010329)

b. Issues and Findings

There were no findings identified. The performance test results showed acceptable heat exchanger performance. However, performance tests PT-2Y10A and PT-2Y10B allowed instrument accuracies of 10% for flow and +/- 1 degree F (Sections 2.5.1 and 2.5.2). Generic Letter (GL) 89-13 allows only 5% for flow and +/- 0.5 degrees F. The performance test results and the heat exchanger performance calculation used the correct instrument accuracies as stated in Con Edison's response to GL 89-13. CR

200010329 was issued to revise PT-2Y10A and PT-210B to reflect the instrument accuracies stated in Con Edison's GL 89-13 response.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspector reviewed Con Edison's follow-up actions for selected structure, system, or component (SSC) issues, to assess the effectiveness of maintenance activities. The inspector reviewed selected risk significant equipment problems that occurred. Items checked included licensee evaluation of functional failures, maintenance preventable functional failures, repetitive failures, availability and reliability monitoring, and system specialist involvement. Additionally, the licensee's Maintenance Rule documents and system condition reports were reviewed and system engineers were interviewed. The auxiliary feedwater system remained in Maintenance Rule Category a(1). The following system/component performance issues were assessed:

- 23 Auxiliary Feedwater Pump Start Failure During Test (CR 200007574)
- 21 Auxiliary Feedwater Pump Shaft Rub (CR 200009054)
- 21 AFW Pump Evaluation - Flowserve Letter dated November 22, 2000
- 21and 23 AFW Pump Vibration Analysis and Performance Trends
- 21 and 23 AFW Pump Oil Sample Analyses

b. Issues and Findings

Condition Report 200007574 described the failure of the 23 AFW pump to start during a surveillance test, and the licensee's actions to evaluate past operability of the pump, correct the deficiency with the alarm switch in the pump supply breaker, and address the extent of condition for this failure mode. Actions taken this period to address the specific deficiency and improve breaker preventive maintenance were acceptable. See also Section 1R21.7 of this report.

Condition report 200009054 described the discovery of mechanical rubs in the 21 AFW pump while investigating bearing lubrication. The 21 AFW pump was returned to the vendor's facility for an overhaul, and was satisfactorily tested and restored to an operable status. The licensee properly evaluated the past operability of the pump, corrected the internal deficiencies, and addressed the extent of condition for this failure mode on the 23 AFW pump. The actions this period to address the specific deficiency and improve auxiliary feedwater reliability were acceptable.

There were no findings identified.

1R13 Maintenance Risk Assessment and Emergent Work

a. Inspection Scope (71111.13)

Throughout the inspection period, the inspectors reviewed the daily and weekly schedules to determine when risk significant activities were scheduled and to assess how the licensee managed risk. The inspectors also discussed selected activities with operations and work control personnel regarding risk evaluations and overall plant configuration control. The inspectors evaluated the licensee's response to the following issues and the risk assessments for inoperable gas turbine (GT) generators.

- Shutdown Risk Assessment for Offsite Power Conditions with High Winds
- Risk Management with Inoperable Gas Turbines (CR 200010357)
- RPS Relay Replacement and Testing (WO 00-18345, 347, 358, 359)
- #23 Cold Leg Thermal Sleeve (CR 200007037, OD 00-16, SE-SQ-12.317)
- NP-99-07456, Preventative Maintenance on 480 volt supply breaker to Station Service Transformer 5

b. Issues and Findings

There were no findings identified. The inspector observed one example where inadequate oversight of the maintenance process resulted in the failure to perform breaker preventive maintenance (PM) as intended, resulting in the need late in the outage to do PM of the supply breakers for Buses 5A and 6A prior to exceeding 350degrees F during plant heatup. The actions to control that maintenance and manage the plant risk were acceptable. Con Edison's decision to not perform the bus work with the plant at power was conservative.

On December 14, 2000, the inspector evaluated the licensee's testing of GT-1 per PT-M38A. While this surveillance test was being performed, GT-3 was removed from service for planned maintenance. During three attempts to start, GT-1 tripped due to stalling and loss of flame. The licensee then declared the turbine inoperable. The inspectors asked operations personnel what the current risk assessment was given that two gas turbine generators were out of service and found that the licensee had not performed such an assessment. When the licensee did calculate the current risk, it was determined that the plant was in a yellow condition (the third highest of four risk categories).

The inspectors verified the licensee's compliance with Technical Specification 3.7.C.1 requiring at least one gas turbine generator and its associated switchgear to be operable. However, the inspectors noted that the licensee had not taken compensatory measures to address the increased plant risk. For example, the licensee did not protect GT-2 as it was the sole operable gas turbine generator at the time; or increase the priority of actions to restore one of the gas turbines to operable status in a timely manner.

The inspector discussed these concerns with outage and maintenance management. The licensee stated that per the outage risk guidance of OAD 38 (Steps 3.1., 6.8 and 6.9), additional contingency actions would be taken for an orange (second highest) risk condition. The licensee reviewed the condition of GT-1 and GT-3 to identify the likely success path for repair, increased the priority of work on GT-3, and restored it to an operable status on December 15, 2000. However, at the end of the inspection period, both GT-1 and GT-2 remained out of service as the licensee addressed ignition basket problems on the turbines.

10 CFR 50.65 a(4) states “Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities.” The licensee stated procedure SAO 161 is being developed to better integrate and align station risk assessments with 10 CFR 50.65 a(4). This item is unresolved pending further NRC review of Con Edison actions to implement 10 CFR 50.65 a(4) **(UNR 05000247/2000-14-01).**

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events

a. Inspection Scope (71111.14)

During the inspection period, the licensee responded to conditions that required operator actions using special or abnormal procedures. The inspectors reviewed the licensee actions for the following:

- AOI 28.0.7 for high wind conditions at the site
- PT-R6, Main Steam Safety Valve Testing

b. Issues and Findings

No significant findings were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspectors reviewed selected operability determinations to assess the adequacy of the evaluations, the use and control of compensatory measures, compliance with the Technical Specifications, and the risk significance of the issues. The inspectors verified that the operability determinations were performed as required by procedure SE-SQ-12.317, Operability Assessments. The inspectors used the Technical Specifications, Technical Requirements Manual, Final Safety Analysis Report, and associated Design Basis Documents as references. The specific issues reviewed included:

- 125 VDC Battery Charger Degraded Voltage and Testing (OD 00-19)
- 22 Station Battery (OD 00-17)
- Pressurizer Code Safety Valve Leakage (OD 00-20)
- RPS Rack Wiring Separation (OD 00-18)
- MS Safety Valve Post-Testing Leakage (CR 200010766)
- Temporary Operator for RHR Valve 1863 (SE-2000-737-TR)

One issue reviewed in detail related to the operability determination in support of a temporary repair of a manually operated, residual heat removal (RHR) system valve. The repaired valve (RHR 1863) is a butterfly valve used in an alternate path for high head recirculation from the RHR pumps to the high head safety injection (SI) system. The normal, post accident and as-failed valve positions were determined and the impact on post-accident and transient conditions was reviewed.

b. Issues and Findings

There were no findings identified.

1R16 Operator Workarounds

a. Inspection Scope (71111.16)

The inspector reviewed and evaluated the licensee's lists of operator work-arounds, control room panel deficiencies, tagouts, operator burdens, and alternate plant configurations. The inspector observed and discussed with control room operators several equipment alignment decisions and actions made in response to performing normal plant equipment manipulations. Several plant transient operations and conditions were walked through with control room operators. A selection of caution tags, temporary modifications, procedure changes, administrative directives, and operator aids were evaluated and discussed with control room operators to identify plant conditions that could affect the operators' abilities to effectively respond to transients and normal equipment operations. The inspector noted the numbers and types of plant deficiencies documented in condition reports and work requests and discussed with shift supervision the cumulative impact these conditions imposed on the ability of operators to respond to plant transients and normal plant manipulations.

b. Issues and Findings

There were no findings identified. Con Edison had established goals to reduce the number of operator burdens and established a goal of having less that 5 workarounds and 5 central control room deficiencies (CCRDIs) prior to restart. The open operator burdens included about 17 CCRDIs and 13 workarounds at the time of startup. The outstanding operator burdens did not significantly affect plant safety and none created an unacceptable condition for plant operation.

1R19 Post Maintenance Testing

a. Inspection Scope (71111.19)

The inspectors observed post-maintenance testing activities and reviewed the IP2 test data. The inspectors verified that the test success criteria addressed in the procedures was in compliance with Technical Specification requirements. The specific issues reviewed included:

- PT–Q17A, Alternate Safe Shutdown Supply to 21 Auxiliary Feedwater Pump
- PT-Q-27A, AFW Pump Quarterly Test
- PMT-187, High Pressure Steam Dump Testing
- PT-M38A, Gas Turbine GT-1 Testing
- PT-R6, Main Steam Safety Valve Lift Setpoint Test
- PT-R75, RCS Leakage Test (CR 200010876)
- GT 27.1-1, 480 Volt Supply Breaker 2022-003
- PMT 19229, Gas Turbine GT-1 Testing
- GT 27.1-1, 480 Volt Supply Breaker 2022-003
- PMT 169321 and 16830, Residual Heat Removal Valve 731

b. Issues and Findings

There were no findings identified.

1R20 Refueling and Outage Activities

.1 Refueling and Restart Preparations

a Inspection Scope (71111.20 and 71715)

The inspectors reviewed the following activities related to the Unit 2 refueling and maintenance outage for conformance to the applicable procedure, and witnessed selected activities associated with each evolution. Operations were reviewed to verify compliance with the technical specification and procedure requirements.

- Shutdown risk evaluations per OAD-38
- Spent fuel pool cooling and operations per SOP 4.3.1
- Operation with the residual heat removal system per SOP 4.2.1
- Industry Operating Experience - VC Summer Pipe Crack Evaluation
- Operating Performance of the 21 RHR Pump
- SOP 17.31, Refueling Surveillance Checklist
- FP-IPP-R15A, Core Reload, TPC 00-0056
- Operation in Reduced Inventory per SOP 1.2 and 4.2.2
- Vacuum Filling and Venting the RCS per SOP 1.1.1

Corrective actions were reviewed for issues described in condition reports (CR) and entered in the corrective action system.

b. Issues and Findings

No significant findings were identified.

.2 Restart Readiness Evaluations and Preparations

a. Inspection Scope (71111.20 and 71715)

The inspector reviewed the licensee's preparations and controls to prepare the plant for startup. The review included a review of the management controls and oversight provided in Temporary Operating Administrative Directive 2 and the Operational Readiness Review Plan. The inspection also included a review of the following activities:

- System Health Evaluations and Presentations
- Revision 36 to the Emergency Operating Procedures (CR 200009552 - SL2)
- Verification of last licensee event report corrective actions (LER 2000-05)
- Reactor Operator Just in Time Training and Lesson Plans
- Startup Challenge 2000- assessment and corrective actions
- Technical Assessment Supporting Heatup 350 degrees F, IPP-00-38
- Control Rod Drop Time Testing per PT-R4A
- PT-V1, Intermediate Range Functional Test
- RCS Leakage Testing per SOP 1.7 and PT-R75
- Supplemental log for accumulator pressure per OAD-3 (CR 2000010697)
- Actions to meet NSAL 00-16 (CR 200003323, 5562)
- Accumulator Pressure Alarm Setpoint Change (SE 00-783-SP)
- Accumulator Temporary Logs (Log DSR-1, CR 200010697)
- Release Permits 00318, 00321, 00325, and 00332
- Plant Heatup Above 350 F per PCO-2
- Precriticality and Reactor Startup per PCO
- Plant Restoration from Cold Shutdown to Hot Shutdown per POP 1.1
- Reactor Startup per POP 1.2
- Plant Startup from Zero Power to Full Power Operation per POP 1.3
- Reactor Approach to Critical at 3:50 am on December 30, 2000

b. Issues and Findings

No significant findings were identified.

.3 Sustained Control Room and Plant Observations of Plant Startup

a. Inspection Scope (71715)

The inspectors implemented an augmented inspection plan during plant startup to permit long-term, heightened observations of control room and plant activities (which included periods of three shift staffing) from December 11 to the end of the period on December 30, 2000, and beyond. Inspectors observed and monitored unit operational activities that implemented a mode change from the cold shutdown condition to the hot shutdown condition. Operator actions to maintain the unit in the hot shutdown condition and in preparation to return to normal temperature and pressure were also observed to ensure that the operators, control room supervision and shift management maintained cognizance of system and component conditions, work activities in the field, and expected off normal plant conditions associated with maintenance and testing activities. Operator actions were observed to ensure that operator response was affected in a timely manner to unexpected plant conditions. Compliance with selected plant technical specifications (TS), plant procedures, and final safety analysis report (FSAR) design basis assumptions were verified.

Plant configurations, equipment status, procedural controls, and compensatory measures consistent with applicable license requirements, design drawings and utility commitments were verified. Control room panel deficiencies, tagouts, work requests, abandoned equipment, alarmed conditions and alternate plant configurations were inspected, validated and discussed with control room operators and supervision. The inspectors discussed recent transient operations with several operators and reviewed recent condition reports involving plant transient response to identify other conditions that could affect the operator's ability to effectively respond to transients. The inspectors also evaluated the cumulative effects of identified conditions on the ability of operators to respond to transients.

The inspectors observed the operation and performance of equipment to verify the adequacy of licensee corrective actions to address deficiencies identified during the February 15, 2000 transient. This review included the performance of the steam jet air ejector pressure control valve, the high pressure steam dumps, the isolation valve seal water system, and the main steam isolation valves. The potential for reactor vessel loose parts was reviewed by direct monitoring of vessel conditions using the digital metal impact monitoring system, and through discussions with operators and contractors. The NRC review of the operability determinations related to loose parts (OD-00-12 and 00-16) was described in NRC Report 05000247/200-013. The inspectors also verified the completion of repairs in the utility tunnel, including the fire water header, pipe supports and other support services important to IP2 operation.

b. Issues and Findings

No significant findings were identified. Inspector observations of performance deficiencies having minor safety significance were discussed with plant management.

1R21 Safety System Design and Performance Capability

.1 Design - Mechanical, Electrical and Instrumentation and Controls

a. Inspection Scope (IP 71111.21)

The team reviewed open items associated with the DC and EDG power supplies including battery 22, battery charger operability, and the EDG loading analysis.

b. Findings and Conclusions

The inspectors reviewed the results of the performance discharge tests performed on the 22 battery, the discharge tests performed at the licensee's third party facility on representative cells, and the material analysis of the disassembled cells performed by the manufacturer and the third party consultants. The inspector reviewed operability determination OD 00-17 and Technical Evaluation Report 00639-TR-001. The inspectors found no issue that would challenge the ability of the 22 battery to power its safety-related design basis loads over the next operating cycle.

The inspectors reviewed the results of the manufacturer's tests of the battery charger's ability to supply its rated load under degraded voltage conditions. The inspector reviewed operability determination OD 00-19. The results of those tests prove the battery charger can successfully operate at 400 Volts or 83% of rated voltage.

The inspectors reviewed the results of the Westinghouse EDG Load Study, forwarded to the licensee by letter LTR-POE-00-142, dated December 18, 2000. No issues were found that would indicate any of the three EDGs would operate beyond its design rating responding to a large break LOCA.

.2 Review of Plant Transient Analysis

a. Inspection Scope (IP 71111.21)

As follow up to the problems identified by the licensee concerning incorrect design inputs to certain safety analyses, the inspectors reviewed the loss of electrical load and turbine trip transient analysis performed by the Westinghouse Corporation, the Indian Point 2 NSSS vendor. The review focused on the source, control, and correctness of the design inputs for: (1) SAS No. 6.0, "Loss of External Load and Turbine Trip," Revision 6, dated October 13, 1999, and (2) Calculation Note CN-TA-88-329, "Indian Point Unit 2 (IPP) Loss of Load/Turbine Trip Analysis For Stretch rating," Revision 2, dated March 1989. The inspectors compared selected key design inputs against the values contained in the Updated Final Safety Analysis Report and the Technical Specifications and discussed them with licensee and Westinghouse engineers.

b. Findings

No findings of significance were identified.

.3 Implementation of Licensing Basis Changes

a. Inspection Scope (IP 71111.21)

The inspectors reviewed Technical Specification (TS) amendment 188, "Best Estimate Large Break LOCA Analysis Methodology," and amendment 211, "Alternate Source Term," to determine whether they were properly implemented in plant procedures. The review included emergency, normal, and abnormal operating procedures, related condition reports and interviews with design and licensing engineers. The inspectors also reviewed portions of an October 8, 1999 Con Edison submittal on the use of the NEI pilot program for implementing the alternate source term methodology for accident analysis as described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." TS amendment 211 used this methodology in implementing various plant equipment modifications.

b. Findings

Prior to August 2000, the licensee did not have a formal process for implementing changes to the plant licensing basis. As a result, certain limits and provisions of TS amendments 188 and 211 were not adequately incorporated into plant operating procedures.

Amendment 188

License amendment 188 was approved by the NRC on March 31, 1997 and implemented by the licensee 30 days later. TS 3.3.A.1.C was changed to incorporate new safety injection accumulator minimum and maximum nitrogen pressures limits of 598 psig and 685 psig, respectively. The new values were the analytical limits contained in the Best Estimate Large Break LOCA Analysis.

In December 2000, the licensee identified a discrepancy between the new TS limits and the minimum allowable pressure value in alarm response procedure ARP SBR-1, "CCR Safeguards" (CR 200009980). Applying pressure instrument loop uncertainty to the procedure's limiting value would have permitted plant operation below the TS limit. The licensee subsequently corrected the procedure.

The inspector identified a similar discrepancy regarding maximum accumulator pressure, in that the licensee did not apply instrument uncertainty to the new maximum pressure TS limit. By applying the instrument uncertainty calculated by the licensee, operation below the ARP alarm setting and within the band permitted by procedure OP 10.1.1, "Safety Injection Accumulators and Refueling Water Storage Tank Operations," could have exceeded the upper TS limit of 685 psig and the accumulator tank relief valve setting of 700 psig. The licensee documented this issue in CR 200010647 and took actions to better assure operation within the pressure limits.

Amendment 211

License amendment 211 was approved by the NRC on July 27, 2000, and implemented by the licensee 30 days later. The submittal and related calculations took credit for 3.4 hours of operation of the containment spray system to remove elemental iodine and particulates from the vapor containment atmosphere. In November 2000, the licensee identified that six different emergency operating procedures still directed operators to stop containment spray after two hours of operation if the containment pressure has been reduced (CR 20000955). One of the acceptance criteria for control room operator doses provided in the submittal was 5 Rem total effective dose equivalent (TEDE). The previous worst case dose calculation considered various ventilation flow rates and resulted in an operator dose of 3.7 TEDE. The inspector discussed the issue with an NRC Region I dose assessment specialist to assess the likely impact of the missed procedure changes on operator dose for the duration of an accident, and concluded that the potential dose exceeded 5.0 TEDE. Thus, premature shutdown of the containment spray system could have caused the acceptance criterion of the analysis to be exceeded.

The licensee’s alternate source term analysis also assumed a reactor coolant system (RCS) dose equivalent iodine-131 limit of 1.0 microcuries per gram. However, TS 3.1.D and its basis were not revised to incorporate the new requirement, and no formal process existed to ensure that the plant was operated within the revised analysis assumptions (CR 20009640). This issue was identified by the licensee and corrected by an established administrative limit prior to the plant reaching a condition in which the TS applied. The licensee intends to incorporate technical specification limits for dose equivalent iodine-131 in a subsequent license amendment.

The findings had a potential credible impact on safety in that: (1) operation outside of the analyzed accumulator pressure band affects fuel peak cladding temperature, and (2) premature cessation of containment spray flow could have adversely affected control room habitability. If left uncorrected, inadequate implementation of license amendments could also result in a more significant safety concern. The inspectors evaluated the findings using phase 1 of the NRC’s significance determination process. The potential for having operated outside of the analytical limits was a design/qualification deficiency that did not affect the operability of the safety injection accumulators per NRC Generic Letter 91-18 (Revision 1), and was a condition of very low safety significance. Premature cessation of containment spray flow following a large break loss of coolant accident was a potential degradation of the radiological barrier function provided for the control room, and was a condition of very low safety significance.

The failure to translate technical specification and design basis requirements associated with the safety injection accumulator pressure, post-accident containment spray system operation, and reactor RCS dose-equivalent iodine into procedures was a violation of Criterion III, “Design Control,” of 10 CFR 50, Appendix B. The inspectors evaluated these conditions using phase one of significance determination process and found them to be of very low safety significance (**Green**). Each issue has been entered into the licensee’s corrective action program as discussed above. This violation was treated as a Non-Cited Violation. (**NCV 05000247/2000-14-02**)

.4 Control of Design Interfaces

a. Inspection Scope (IP 71111.21)

The inspectors reviewed the circumstances related to inconsistences that were identified by the licensee in input data that was used by the NSSS vendor to perform Indian Point 2 accident analyses. The design inputs primarily affected the best estimate loss of coolant accident (LOCA) analysis and post-LOCA containment pressure response. Related technical issues contained in the following licensee condition reports were evaluated:

- 200008958 - Non-conservative value for RHR flow in containment peak pressure calculation
- 200000035 - RWST levels inconsistently applied in accident analyses
- 200009750 - Motor-operated valve gear changes increased stroke times used in containment peak pressure analysis
- 200009752 - Different diesel generator start times between FSAR and accident analysis
- 200001738 - Flow coefficient values for accumulator injection and pressurizer surge lines do not agree with as-built plant condition

b. Findings

The licensee does not have formal design procedures to control the verification, validation and supply of input data and assumptions to the NSSS vendor. Procedure controls were inadequate to ensure that input assumptions that were used in the accident analyses were not invalidated by plant modifications. Aspects of this finding that pertain to performance in the area of problem identification and correction are discussed in Section 4OA2.

Design inputs to the accident analyses were not always formally verified, validated, and transmitted to the NSSS vendor. Instances were identified in which design inputs were communicated or changed by means of telephone conversations, electronic mail between individuals, or by hand markup in the margins of memoranda. As a result, discrepancies were created between the values assumed in the analyses and actual plant conditions. The accident analyses input assumption discrepancies that the inspectors reviewed had potential impact on post-LOCA fuel peak cladding temperature (PCT) and containment peak pressure by directly or indirectly affecting mass-energy produced and released to the containment. After identifying the discrepancies, the licensee requested the NSSS vendor to re-perform the accident analyses using verified and correct design inputs. By eliminating excess conservatism from the containment integrity analysis, the calculated peak containment pressure was reduced from 44.45 psig to 43.0 psig (The design basis limit is 47.0 psig.). Fuel PCT increased approximately 4.0 degrees Fahrenheit to 2183°F. The limit on fuel PCT is 2200 degrees Fahrenheit. Therefore, the collective affect of the discrepancies on the plant design basis was minimal.

However, the lack of formal controls of design inputs, if left uncorrected, could become a more significant concern and could affect the integrity of the fuel cladding or the reactor containment boundary. The inspectors evaluated lack of controls for design inputs using phase 1 of the NRC significance determination process and characterized the condition as having very low safety significance. (**Green**)

10 CFR 50, Appendix B, Criterion III, “Design Control,” requires measures to be established for the identification and control of design interfaces and for coordination among participating design organizations. The Con Edison Quality Assurance Program Description commits to conform to Regulatory Guide (RG) 1.64, which, in turn, endorses ANSI Standard N45.2-11-1974, “Quality Assurance Requirements for the Design of Nuclear Power Plants.” Section 5.0 of the ANSI standard requires that the external interfaces between organizations performing work affecting the quality of design shall be identified in writing, that systematic methods shall be established for communicating needed design information across external design interfaces, including changes to the design information as work progresses, and that procedures shall be established to control the flow of design information between organizations.

Contrary to the above, the licensee did not establish measures for the identification and control of design interfaces and for coordination among participating design organizations as described in ANSI N45.2-11-1974. As a result, discrepancies between actual plant conditions and input assumptions in the accident analyses existed that affected design margins related to the fuel and containment barriers. However, because of the very low safety significance of the specific input discrepancies and because the licensee has included this item in its corrective action program (CR 200009093), this violation was treated as a Non-Cited Violation. (**NCV 05000247/2000-14-03**)

.5 Corrective Actions for Control of Vendor Interfaces

a. Inspection Scope (IP 71111.21)

The team reviewed a sample of revised design input parameters used in accident analyses which had been identified by the licensee’s NSSS vendor as incorrect. This review included calculations, drawings and design inputs associated with RHR flow, and the available refueling water storage tank (RWST) inventory.

1. Findings

The licensee NSSS vendor had evaluated the impact of the design input errors discussed below and determined that the affect on the maximum peak containment pressure was not significant. However, the NSSS vendor had preliminarily determined that the combined affects of these errors would result in a slightly higher peak fuel clad temperature, which remained below the regulatory limits established in 10 CFR 50.46. Licensee corrective actions were in progress at the conclusion of this inspection, which included reviews to assure prior to startup that the plant was operated within required limits.

b.1 Minimum Deliverable RWST Volume Design Inputs

The licensee and the NSSS vendor had identified that the current containment analysis used design input assumptions for the minimum deliverable volumes available from the RWST which were not consistent with the plant licensing basis. The NSSS vendor had used 250,000 gallons deliverable at the low level alarm and 80,000 additional gallons at the lo-lo level alarm as design inputs for the loss of coolant accident mass and energy release analysis which determined the maximum containment pressure increase. Based on the RWST instrument uncertainties and the minimum required RWST level required by Technical Specifications, the licensee determined that the correct values for these design inputs should be 229,000 gallons and 68,000 gallons. The inspector confirmed that these input values were appropriate and conservative. However, the licensee did not have a formal calculation which explicitly determined these new design input values. Licensee actions were in progress at the end of the inspection to perform a formal calculation for these inputs.

During review of this issue, the inspector identified three minor design related discrepancies associated with the RWST as discussed below.

(1) The inspector could not confirm the available tank vent area based on the drawing for the RWST vent. The RWST vent drawing showed a perforated plate which covered the vent path, but did not provide details of the available flow area. If sufficient vent area was not available, a partial vacuum would be drawn inside the tank during emergency core cooling system operation, which would affect the indicated tank level and adversely challenge operators. Licensee actions were in progress at the end of the inspection to confirm that an adequate vent area existed.

(2) The inspector identified that the licensee lacked calculations to demonstrate the seismic adequacy of the internal overflow vent line for the RWST. This internal overflow pipe was 12 inches in diameter, 36 feet tall and penetrated the wall of the tank near the bottom of the tank. The overflow pipe was supported at the penetration of the tank wall with a fillet weld and by lateral U-bolt restraints attached to 3 inch angle beams fillet welded to the tank wall at two locations. With this configuration, the team questioned if the tank wall would fail under seismic loads at the supporting attachment welds causing a tank leak and loss of inventory. The licensee intended to investigate this concern. This issue was considered minor because sufficient alternate makeup water supply was available for plant shutdown following a seismic event.

(3) The inspector identified that emergency operating procedure ES 1.3 “Transfer to Cold Leg Recirculation,” had established a nonconservative RWST tank level for securing the containment spray pumps. Step 35 required securing the containment spray pumps after RWST level decreased below 2.0 feet. The team calculated a minimum level required to prevent air entrainment for these pumps of 2.25 feet. This was based on a 1.49 foot tank level required for minimum submergence and a 0.76 foot level allowance needed for instrument uncertainty. The licensee acknowledged the potential for air entrainment and pump damage, but stated that at this point in the accident sequence, the core spray pumps were no longer needed to support accident mitigation.

Overall, considering these minor design deficiencies, the licensee determined the system was still operable. The NRC team similarly considered the issues minor, however, these items are unresolved pending completion of the licensee evaluations and subsequent review by the NRC. **(UNR 05000247/2000-14-04)**

.6 Safety Injection System Evaluation

a. Inspection Scope (IP 71111.21)

The inspectors evaluated the condition of the safety injection (SI) system to determine system readiness. The inspectors walked down the system and performed the following:

- Inspected the material condition of SI system equipment both inside and outside containment.
- Reviewed the technical adequacy of selected SI-related procedures, including station operating procedures (SOPs), abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and alarm response procedures (ARPs).
- Interviewed operators, system engineers and Con Edison management personnel regarding the condition of the SI system.
- Attended the system readiness review presentation for the SI system.
- Reviewed open and closed operability determinations completed over the last two years.
- Reviewed selected surveillance tests to confirm that critical design parameters were appropriately incorporated.
- Reviewed selected surveillance tests to verify that the test results were within the specified acceptance criteria.
- Reviewed the status of preventive maintenance tasks, and verified that deferred tasks were adequately justified.
- Reviewed a list of open temporary facility changes, “operator work-a-rounds” and control room deficiencies. Based on this review, specific items were selected for detailed review.
- Reviewed lists of open and closed work orders, condition reports (CRs), and requests for engineering services (RESs) associated with the SI system. Based on this review several items were selected for a detailed review.
- Reviewed the work history of the accumulator level and pressure instruments and the refueling water storage tank (RWST) level instruments to determine if Con Edison had fixed previously identified problems.
- Reviewed the open work orders, condition reports and engineering service requests to confirm that significant items were scheduled to be completed prior to plant startup.
- Reviewed Con Edison’s response to selected information notices and other industry information associated with the SI system.

- Observed performance and reviewed documentation associated with SI system performance test Q29-A, “21 Safety Injection Pump Test.”
- Reviewed the items associated with the SI system that Con Edison decided not to complete prior to startup and evaluated the cumulative impact of these items on the system.

b. Findings

The inspectors identified no conditions that affected the SI system operability. In addition, the inspectors identified no concerns indicating that the SI system was not ready for a plant restart provided that Con Edison completed the SI system-related startup hold items as planned. However, the inspectors identified that no corrective actions were taken to prevent recurrence of material condition concerns with the freeze protection for the RWST, primary water storage tank (PWST) and condensate storage tank (CST) level switches. Over the last three years several CRs associated with the material condition of the freeze protection for these level switches had been generated, some of which were associated with actual failures of the switches. Although in each case corrective actions were taken to address the specific failure, no corrective actions were taken to prevent recurrence of problems with the freeze protection of these level instruments. This finding was evaluated using Phase 1 of the significance determination process (SDP) and determined to be **Green** (of very low safety significance) since it did not result in the actual loss of a safety function.

The RWST, PWST and CST level instruments provide alarm in the control rooms that require the operators to take certain actions. Most noteworthy is the RWST low level alarm; this alarm is specified in the EOPs for a large-break loss-of-coolant-accident (LOCA), and requires the operators to swap the suction for the SI pumps from the RWST to the recirculation sump, to ensure adequate core cooling.

As described in the Updated Final Safety Analysis Report, the RWST, PWST and CST level instruments are provided with freeze protection. The Electric Heat Tracing system description specifies that freeze protection is provided by the use of electrical heat tracing in conjunction with insulation such that the items being protected will be maintained within a temperature range of 35 degrees Fahrenheit (^{o}F) to 45^{o}F with ambient temperatures of -10^{o}F and a wind velocity of 40 miles-per-hour (mph). Based on a review of the work history and past CRs for these level instruments several material condition deficiencies have been documented, some that resulted in actual failures of the level switches. These deficiencies include such things as damaged insulation and improperly attached electrical heat trace. Although Con Edison had taken corrective actions to address each specific deficiency, these problems reoccurred. Based on the inspector’s review, no corrective actions were established to prevent recurrence of these problems with the freeze protection for the level switches.

The failure to take corrective actions to preclude repetition was a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," which requires actions to be taken to preclude repetition of significant conditions adverse to quality. This failure to take corrective action to prevent repetition was determined to be of very low safety significance **(Green)**. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). The issues associated with this violation are in the Con Edison corrective action system as CR 200009661. **(NCV 05000247/2000-014-05)**

.7 Auxiliary Feedwater Pump Failure to Start

1. Inspection Scope (IP 71111.21)

The inspector reviewed a restart item associated with the 23 auxiliary feedwater (AFW) pump failure to start on demand. The issue involved the failure of an alarm switch contact in the closing circuit of the pump supply circuit breaker, a Westinghouse type DB-50. The switch contact, that is normally closed and opens on overcurrent, failed open due to aging. The failure of the contact prevented the breaker from closing and the pump from starting.

The inspector's review addressed the failure itself, the licensee's immediate corrective actions, and the actions taken to evaluate and address extent of condition. Specifically, the inspector reviewed the applicable control wiring diagram, Drawing No. 9321-LL-3118-29, Sheet 16, dated July 14, 2000, the original condition report, No. 20007618, and the revised version of the same report, dated December 12, 2000. The inspector also evaluated: the results of the initial investigation and subsequent root cause analysis; selected portions of the breaker preventive maintenance (PM) procedure) No BRK-P-002-A, Revision 3), and the results of the last performance tests conducted on the failed circuit breaker; the revisions made to the PM procedure and the results of improved PM tests conducted on the affected breakers; and the licensee's actions taken to confirm operability of the affected breakers. The inspector also evaluated past licensee experience with the same breakers and conducted interviews with the system engineer to verify the correctness of the documents reviewed.

b. Findings and Conclusions

There were no significant findings identified with this issue.

.8 RPS Logic Protection Rack Wiring Separation

1. Inspection Scope (IP 71111.21)

The team reviewed a restart item associated with the reactor protection system (RPS) logic protection rack wiring separation concerns. The issue involved the licensee's observation that wiring within the protection racks did not always conform with the statements contained in the UFSAR and the installation and electrical separation criteria contained in drawing A208685. Specifically, the licensee found instances of wires associated with computer and alarm circuits being in close proximity of, and some times in the same cable tray as, the wires associated with the trip and logic circuits. The licensee also identified examples of switch contacts originally reserved for logic and trip function being used for computer and alarm functions. All potential interactions involved a single train of protection logic and low energy and low voltage circuits.

The inspector reviewed three condition reports related to this issue (Nos. 200007597, 200008818, and 200009499), and the associated Operability Determination (OD), No. 00-018, Revision 0, dated November 28, 2000. The inspector also reviewed the OD supporting documentation, including a Westinghouse letter, dated November 22, 2000, addressing the original separation requirements of the racks, and Con Edison's calculation No. FEX-00146-00, Revision 0, addressing circuit protection and fuse coordination.

b. Findings and Conclusions

There were no significant findings identified with this issue.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspectors observed the performance of selected portions of surveillance tests and reviewed portions of the test results to verify that the tested systems and components were capable of performing their safety functions including:

- PT-Q27A. Quarterly flow surveillance on the 21 Auxiliary Feed Pump
- SOP 1.7, RCS Leakage Surveillance
- PT-R4A, Control Rod Drop Time Testing
- PT-R53A, RHR Isolation Valve 730 and 731 Testing
- PTV-21, Low Head Injection and RHR Check Valve Test
- PT-Q26A, 21 Service Water Pump
- RFE-S-16.032, Zero Power Physics Testing
- PT-A36A, Gas Turbine 1 Overspeed Testing
- PT-R6, MS Safety Valve Test Methodology and Leakage (CR 200010766)
- PT-R75, RCS Integrity Inspection

b. Issues and Findings

The inspector noted two deficiencies regarding AFW Pump Testing per PT-Q27A regarding the test method to verify check valves BFD-52 and BFD-54 (Condition Report 20010066); and, the accuracy flow instrumentation as required by ASME Boiler and Pressure Vessel Code Section XI of the(Condition Report 200009754).

In regard to PT-Q27A, Con Edison does not require a new reference value to be established or the previous values reconfirmed prior to declaring the pump operable. Indian Point 2 Technical Specification 4.2.1, “Inservice Testing,” requires that inservice testing of pumps whose function is required for safety shall be performed in accordance with the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code. OMa-1988, Part 6, delineates the requirements for pump testing. Specifically, section 4.4, “Effect of Pump replacement, Repair, and Maintenance on reference Values,” states, in part that, when a reference value may have been affected by repair, a new reference value or set of values shall be determined or the previous value reconfirmed by an inservice test run prior to declaring the pump operable. Con Ed personnel stated this review is performed by the engineering review to set the test procedure acceptance criteria prior to conducting the test, through the use of “minimum engineering values” for certain parameters. There are no additional reviews that are required after major pump maintenance to verify or reestablish new reference valves prior to declaring the pump operable. However, in this case the pump was not declared operable due to additional refueling outage testing that must be completed. The engineering evaluation was subsequently completed prior to declaring the pump operable.

No significant findings were identified. The inspector also noted several questions regarding the testing of the main steam safety valves per PT-R6. Specifically, the inspector questioned: accounting for a +/- 1% statistical correlation error between instrument lift setpoint and actual valve pop setpoint in the valve lift setpoint determination; acceptability of post-testing weepage past the valve seats (CR 200010766); accounting for differences in ambient room temperature (105F vs 75F) in the test results; verification of valve seat area; acceptance and validation of vendor test methods; accounting of 0.5% drift error from post-test equipment calibration in test results.

Together the issues regarding PT-Q27A and PT-R7 are open pending further NRC review of testing under the ASME Code. **(UNR 05000247/2000-14-06)**

1R23 <u>Temporary Plant Modifications</u>

a. <u>Inspection Scope</u> (71111.23)

The inspector reviewed the following temporary plant modifications:

* TFC 00-276, Temporary Secondary Plant Makeup Supply
* SE 2000-737-TR, Temporary Operator for RHR Valve 1863

b. Issues and Findings

Temporary Facility Change (TFC) 00-276/SE-97-29-TM was issued to provide a temporary supply of makeup water to support plant startup. The TFC used canvas hoses to supply filtered water from the fire system to Unit 1 storage tanks. The fire hoses ran through the Unit 1 switchyard containing 138 KV bus bars and breakers. The temporary water supply was not implemented consistent with the TFC safety evaluation. The licensee isolated the makeup supply until the following actions were taken: restraints and rubber pads were added to secure the hose and prevent fretting in locations where it passed near the 138KV bus bars; access postings and personnel barricades were improved to prevent inadvertent entry to the area; and, the nuclear plant operators monitored the status of the TFC each shift. The failure to implement this TFC in accordance with the safety evaluation was an example of a recurrent problem at IP2. The licensee entered this matter in the corrective action program as Condition Report 200010418.

The inspector reviewed the summary descriptions of several recent temporary repairs (TR) and a listing of the current TR in place in the field. A detailed evaluation was conducted of temporary repair SE 2000-737-TR, which was performed on a residual heat removal (RHR) system valve (RHR 1863). Valve RHR 1863 is a manually operated butterfly valve whose operator was found separated from the valve stem. The adequacy of the repair, which involved squaring and severing the existing damaged valve stem and attaching a face plate to the valve body with equal threaded fasteners, was determined. The inspector discussed the specific impact of the RHR valve modification and the cumulative impact of the temporary modifications currently in place with control room operators and supervisors.

There were no significant findings.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Review

.1 Performance Indicator Data Collecting and Reporting

a. Inspection Scope (TI 2515/114)

The inspector reviewed the licensee's performance indicator data collecting and reporting process as described in procedure SAO-114, "Preparation of NRC and WANO Performance Indicators". The purpose of the review was to determine whether the methods for reporting PI data are consistent with the guidance contained in NEI 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guidelines." The inspection included a review of the indicator definitions, data reporting elements, calculational methods, definition of terms, and clarifying notes for the security equipment performance indicator. Past NRC reviews of PI data collecting and reporting were documented in NRC Reports 05000247/2000-01, -06, -08, -09 and 11.

b. Issues and Findings

No significant findings were identified.

4OA2 Cross Cutting Issues

.1 Corrective Actions for Problems Concerning Control of External Design Interfaces

a. Inspection Scope

The inspectors reviewed condition reports 200009093, "Apparent lack of control over accident analysis inputs," and 200009666, "Improper closure of NRC inspection item 50-247/98-201-13," to assess the licensee's performance in the area of corrective action for similar, previously identified issues. The inspector discussed the issue with members of the licensee's root cause assessment team and selected engineering staff, and reviewed the following documents:

- Con Edison Nuclear Quality Assurance Audit Report No. 97-08-D, "Design Control: Design Bases"
- NRC Inspection Report Nos. 50-247/98-201 and 50-247/98-08
- NRC Information Notice 98-22, "Deficiencies Identified During NRC Design Inspections"
- Indian Point Unit 2 Quality Assurance Program Description, Revision 15
- Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants," Revision 2
- ANSI N45.2-11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

b. Findings

The violation of the design interface control requirements that is discussed in Section 1R21.3 of this report was a condition adverse to quality that had been identified previously by the licensee and the NRC.

Quality Assurance Audit Finding 97-08-D-F03, dated November 14, 1997, stated that external interfaces between Nuclear Safety and Licensing and contracted engineering organizations performing work affecting the quality of design had not been identified in writing, and that procedures had not been established to control the flow of design information from Con Edison organizations to the external groups and back. There were no procedure controls to assure that the design information transmitted from one organization to the other was properly documented, uniquely identified, or issued by authorized persons. CR 199803209 for this finding was closed in November 1999.

NRC Design Inspection 50-247/98-201, Section E1.3.2.2.(k), documented that the licensee had not established specific procedures or controls for acquiring, verifying, transmitting, reviewing, or controlling plant specific inputs provided to Westinghouse Corporation for a large break loss of coolant accident analysis. This observation subsequently was summarized in NRC Information Notice 98-22, dated June 7, 1998. In NRC Inspection report 05000247/1998-08, closed the observation on the basis of a reorganization that placed the nuclear safety and licensing group into the nuclear design organization, subject to the procedures in the Engineering Operations Manual, and a plan for engineering to develop a design basis/accident analysis parameter “roadmap” to facilitate access to design basis information. The responsibility to develop the “roadmap” was later given to the configuration management organization. As of this inspection, software for an accident analysis input database had been developed, but limited population of the database has been funded in the Indian Point 2 Business Plan for the Year 2001. Thus, corrective actions for the previous findings were incomplete at the time of this inspection.

As an interim corrective action, the IP2 Chief Nuclear Engineer issued a directive that established formal administrative control of the interface between ConEd and the NSSS vendor. The directive required verification and validation of all design inputs, and designated the method and personnel that were authorized to transmit design information.

10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requires measures to be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this requirement, conditions adverse to quality involving a violation of the design interface control requirements of 10 CFR 50, Appendix B, Criterion III that were previously identified in 1997 and 1998 by the licensee and the NRC were not promptly corrected. As a result, multiple discrepancies existed between design inputs used in accident analyses and actual plant conditions. The issue has a credible impact on safety due to its potential affect on safety margins, which left uncorrected could become a more significant safety concern. This violation was treated as a Non-Cited Violation (**No Color**), consistent with Section VI.1.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368), because of : (1) the very low safety significance of the condition adverse to quality described in Section 1R21.3 of this report; (2) the adequacy of the licensee’s interim corrective actions in response to CR 200009093; and (3) the item being included in the licensee’s corrective action program as CRs 200009093 and 200009666. (**NCV 05000247/2000-14-07**)

4OA3 Steam Generator Replacement Project (500001)

a. Inspection Scope (IP50001)

This inspection included a review of five sets of radiographs representing the reactor coolant system (RCS), main steam and feedwater welds, and the weld documentation process. The radiographs for the following work packages were reviewed:

- SG#23, WP 3065C, Weld FW-1-AA, RCS Hot Leg to Nozzle
- SG# 23, WP 3065C, Weld FW-2-AA, RCS Cold Leg to Nozzle
- SG#24, WP 3085D, Weld FW-1-AA, 18 inch Feedwater pipe to nozzle
- SG#24, WP 3030D, Weld FW-1-AA, 31 inch Main Steam pipe to nozzle (repairs)
- SG#24, WP 3065D, Weld FW-2-AA, RCS Cold Leg to Nozzle,

This review was performed to verify there were no unacceptable conditions in the final welds, and that all relevant radiograph indications were properly dispositioned.

b. Observations and Findings

No findings were identified.

4OA4 Human Performance

a. Inspection Scope

The inspectors observed personnel performance during refueling and startup activities. In addition, the inspectors reviewed corrective actions for errors occurring during the period.

b. Issues and Findings

The inspectors had several observations that were indicative of inadequate procedures or personnel performance. All the issues had minor safety significance. Examples included:

- The failure to install a seal on valve SWN-642, the 25 Fan Cooler Unit Inlet Vent Stop. SWN-642 was in the correct position (CR 200010510).

- The failure to restore the 25 service water pump power supply from Bus 3A as intended prior to exceeding 200F during the RCS heatup. The NRC identified the supply breaker was not restored with the RCS at 330 F. The 3A supply was available prior to 350 F, as required by technical specifications. However, during restoration, Con Edison initially found that they failed to perform a post-maintenance test on the breaker (CR 200010516).

- The licensee's evaluation of the need to revise PT-Q26A to address the need for operators to isolate service water chlorination, and discuss controls for confined space entry into the lower pump areas (CR 200010760).

- The licensee's evaluation of a condition identified during overspeed testing of GT-1 that the test method and overlap of the mechanical and electrical overspeed setpoints can result in circumstances where the mechanical overspeed is not checked as intended (CR 200010230).

No significant findings were identified. The licensee addressed the issues in the condition reports referenced above.

4OA5 Inspection Followup Items

(Closed) LER 05000247/2000-005: Steam Generator Design Differential Pressure. The inspector reviewed the licensee actions in response to this issue for the old stream generators. The corrective actions were appropriate to update the design specification and stress reports per ASME, Section XI. This item is closed.

4OA6 Meetings

Exit Meeting Summary

On January 25, 2001, the resident inspectors presented the inspection results to Mr. A. Blind and other members of the Con Edison staff who acknowledged the findings. The inspectors obtained proprietary information during the review of safety system performance, which was returned to the licensee at the conclusion of the inspection.

4OA7 Licensee Identified Violations

The following findings of very low significance were identified by IP2 and are violations of NRC requirements which meet Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-Cited Violations (NCVs).

a. ***NCV 05000247/2000-14-08*** 10CFR 73.21(a), Requirements for the protection of safeguards information requires, in part, "Each licensee....shall ensure that Safeguards Information is protected against unauthorized disclosure." In September, 2000, the improper handling of Safeguards documents was identified; as described in the licensee corrective action program, Reference Condition report 200007569.

b. ***NCV 05000247/2000-14-09*** 10CFR 26 Appendix A, Failure to Implement Requirements for FFD Testing. QA Annual Audit 00-04-D of the Fitness for Duty (FFD) Program identified that samples sent to the offsite lab for analysis were not tested to the correct criteria. Followup actions were appropriate. Reference Condition Report 200009066.

ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

05000247/2000-13	LER	Steam Generator Design Pressure Rating

Opened

05000247/2000-14-01	UNR	Actions to Assess and Manage Risk During Maintenance
05000247/2000-14-04	UNR	Evaluation of RWST Design
05000247/2000-14-06	UNR	Evaluation of Testing Under Section XI

Opened and Closed During this Inspection

05000247/2000-14-02	NCV	Failure to translate the design basis into procedures
05000247/2000-14-03	NCV	Failure to control of design interfaces
05000247/2000-14-05	NCV	Failure to take corrective actions for freeze protection
05000247/2000-14-07	NCV	Failure to correct inadequate design interfaces
05000247/2000-14-08	NCV	Failure to control safeguards information
05000247/2000-14-09	NCV	Failure to conduct adequate FFD testing

LIST OF DOCUMENTS REVIEWED

Condition Reports

199805570, Westinghouse NSAL-98-004 pipe schedules and accident analyses, 6/29/1998
199908742, Westinghouse NSAL-98-004 pipe schedules and accident analyses, 11/19/1999
200001738, Best Estimate LOCA analysis as-built pipe schedule inputs, 3/13/2000
200004562, As-built pipe schedule impact on 10CFR 50.46 limits, 6/15/2000
200004333, SI Accumulator Relief Valve 892C leaks by, 6/7/2000
200008958, Containment analysis does not reflect RHR flows, 11/13/2000
200009642, EOP ES-1.3 assumes fan cooler unit charcoal filters are installed, 11/30/2000
200009645, Containment peak pressure calculation used nonconservative inputs, 10/ 6/2000
200009750, Valve stroke times greater than assumed in containment analysis, 12/2/2000
200009639, Increased post-LOCA containment fan cooler unit heat removal rates, 11/30/2000
200009644, Error in containment spray flow calculation, 11/30/2000
200009640, TS 3.1.D.1 RCS specific activity may not bound analysis assumptions, 11/30/2000
200005392, EDG load study safety evaluation conditions not met, 7/19/2000
200009759, Inadequate Recirculation pump NPSH calculation 12/2/2000
200009374, Procedure PT-R139 contained incorrect acceptance criterion, 11/23/2000
200009552, EOPs inconsistent with TS Amendment 211 assumptions, 11/28/2000
200009643, Incorrect assumptions for accumulators in containment calculation, 11/30/2000
200009666, Procedures for control of inputs to analyses were not developed, 12/1/2000
200009752, Incomplete reviews of nuclear safety evaluation 98-402-PR, 12/2/2000
1998-0309, QA Design basis audit findings and recommendations, 12/24/1997
200010647, Accumulator instrument uncertainty disagreement, 12/21/2000
200010074, Accumulator low pressure in technical specifications and procedures, 12/8/2000
200009980, Discrepancy in accumulator pressure in technical specifications, 12/6/2000
200009093, Inadequate control over accident analysis inputs, 11/20/2000
199702183, PT-V13 failed to meet acceptance criteria on 23 SI Pump, 6/6/1997
200000340, Frozen transmitter for the condensate storage tank (CST), 1/17/2000
200000352, Adequacy of design of support equipment for winterization protection, 1/18/2000
200000363, Primary water storage tank (PWST) level alarm due to extreme cold, 1/18/2000

200000374, PWST frozen level instrument sensing line, 1/18/2000
200000381, Material condition in PWST instrument enclosure and freeze protection, 1/19/2000
200000382, Recent failures of level instruments due to extreme cold weather, 1/19/2000
200005455, Bearing in 22 SI Pump may not be environmentally qualified, 7/21/2000

Procedures

- ARG-2, Shutdown LOCA, Revision 1, dated September 30, 1997
- AOI 4.2.2, LOCA When RCS Temperature At Least 200°F And Less Than 350°F, Revision 4, dated April 5, 2000
- ES-1.3, Transfer to Cold Leg Recirculation, Revision 35 and draft Revision 36
- SAO 451, Verification, Documentation, and Traceability of Calculations, Revision 5, dated April 11, 2000
- SAO-139, Process For Control Of Changes To The Updated Final Safety Analysis Report (UFSAR), Revision 0, dated December 16, 1998
- SAO-465, License Amendment Requests (LAR), Revision 0, dated August 25, 2000
- DSR-1, Unit 2 Central Control Room Log, Revision 74, dated December 19, 2000
- SOP 10.1.1, Safety Injection Accumulators And Refueling Water Storage Tank Operations, Revision 33, dated December 19, 2000
- ARP SBF-1, CCR Safeguards, Revision 22, dated December 19, 2000
- DE-SQ-12.513, Design Verification, Revision 0, dated June 15, 1998
- Engineering Operations Manual, Section 5.16, Preparation and Review of Design and Engineering Analyses, dated April 28, 1998
- Engineering Operations Manual, Section 5.18, Control of Engineering Contractors, dated September 30, 1999
- SAO-112, Corrective Action Program, Revision 3

Drawings

- B235414-01, Stress Isometric For Piping Problem SI-351, dated September 3, 1998
- B235335-00, Stress Isometric For Piping Problem SI-352, dated August 1, 1993
- B235336-01, Stress Isometric For Piping Problem SI-353 ½, dated September 3, 1998
- 20339-OR, “Refueling Water Storage Tank” Revision 1.
- 206103-4, “Installation of Alarm Switch on Refueling Water Storage Tank” Revision 4.
- 201262, “Refueling Water Storage Tank Overflow Piping” Revision 5.
- 20339-20, “12 inch Shell Overflow Nozzle E”, Revision 0.

Miscellaneous Documents

- Westinghouse Nuclear Safety Advisory Letter (NSAL)-98-004, Accumulator Injection/Surge Line Piping Parameters, dated June 19, 1998
- Consolidated Edison Company Letter NL 00-032, Reply to Notice of Violation and Proposed Imposition of Civil Penalty, dated March 27, 2000
- Nuclear Quality Assurance Audit Report 97-08-D, Design Control: Design Bases, dated November 1997
- RHR/SIS Design Basis Document, Revision 0
- Safety Evaluation No. 00-439-SP, Accumulator Pressure Low Alarm Setpoint Change, Revision 00, dated July 20, 2000

- Consolidated Edison Company Letter, Request For Plant Specific Approval Of Best Estimate Large Break LOCA Analysis and Proposed Changes To Technical Specifications To Utilize For Indian Point Unit No. 2, dated August 14, 1996
- Indian Point 2 Maintenance Rule Basis Document - Safety Injection System
- RHR/SIS Design Bases Document (DBD), Rev. 0 - Residual Heat Removal/Safety Injection System
- System Description No. 30: Electric Heat Tracing
- Safety Evaluation SE 99-267 MD - Accumulator level setpoint change
- Technical Specification Sections: 3.3, 3.6, and 6.0
- Emergency Operating Procedures: E-0, E-1, ES 1.3
- Consolidated Edison Letter to Westinghouse, “Consolidated Edison Company of New York Indian Point 2 Verification of Input Data,” dated December 2, 2000.

Surveillance Tests

PT-Q29A, Rev. 12	21 Safety Injection Pump Test
PT-Q29B, Rev. 12	22 Safety Injection Pump Test
PT-Q29C, Rev. 12	23 Safety Injection Pump Test
PT-R16, Rev. 13	Recirculation Pumps
PC-R21B-1, Rev. 5	Recirculation Sump Continuous Level Transmitter
PT-V13, Rev. 6	Safety Injection System Hydraulic Functional
PC-Q2, Rev. 14	Refueling Water Storage Tank Level
PC-R-9-1, Rev. 4	RHR System Flow Transmitters
PC-A-1, Rev. 4	Accumulator Pressure CCR

Work Orders

NP-00-18293 PM - Re-grease inboard and outboard motor bearings per lube sheet SI-L02

System Readiness/Health Reports

Safety Injection including RWST and Containment Recirculation, dated November 19, 2000
Electric Heat Trace System, dated February 12, 1999

Operability Determination

OD 99-006	Allowable Error RWST Level Indication
OD-00-004	SI Accumulators
OD-00-015	22 RHR pump vibration

Temporary Facility Changes

2000-251	Temporary Enclosure and Heat Lamp for RWST Level instruments

Licensee response to the following OE

Information Notice 97-60, “Incorrect Unreviewed Safety Question Determination Related to Emergency Core Cooling System Swapover.” CR 199703012
SOER 97-1, Potential gas binding of SI and charging pumps CR 1997047409
Information Notice 88-03, Safety Injection Pipe Failure
Information Notice 93-13, Undedicated modification of flow characteristics in HPSI system

Calculations:

MPN-S65-001 “RWST- Level Instrument Accuracies Calibration and Setpoints” Revision 1.
FMX-00036-04 “SI-Recirculation Pump Available NPSH” Revision 4.
FMX-00085-01 “RWST Minimum Submergence Level” Revision 0.

PARTIAL LIST OF INDIVIDUALS CONTACTED

L. Guercio	Security Superintendent
T. Elsroth	Security Supervisor
B. Allen	Regulatory Affairs Manager
P. Cordero	System Engineer, Electric Heat Tracing System
J. Ferrick	Operations Manager
R. Masse	Plant Manager
T. McCaffery	Plant Engineering
J. McCann	Licensing Manager
G. Schwartz	Chief Engineer
D. Shah	System Engineer, Safety Injection System
A. Spaziani	Compliance Engineer
M. Vassely	System Engineering Section Manager
J. Ventosa	Site Engineering Manager

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
ARM	Area Radiation Monitor
ARP	Alarm Response Procedure
CCTV	Closed Circuit Television Camera
CCR	Central Control Room
CFR	Code of Federal Regulations
CR	Condition Report
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ERO	Emergency Response Organization
ESW	Emergency Service Water
FR	Federal Register
FSAR	Final Safety Analysis Report
HP	Health Physics
IDS	Intrusion Detection System
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PASS	Post Accident Sampling System
PCT	Peak Cladding Temperature
PMT	Post Maintenance Test
PI	Performance Indicator
psig	pounds per square inch, gage
QA	Quality Assurance
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SCBA	Self-Contained Breathing Apparatus
SRV	Safety Relief Valves
SGTS	Standby Gas Treatment System
SSC	Structure, System, or Component
TEDE	Total Effective Dose Equivalent
TIP	Transverse In-Core Probe
TS	Technical Specification

ATTACHMENT I

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revised 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), radiation safety (protecting plant employees and the public), 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 cornerstones of safety, the NRC uses inspections and performance indicators that generate information about the safety significance of plant operations. 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 NRC can reach objective conclusions regarding overall plant performance. The NRC 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 actions in response to the significance 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.