October 26, 2005

Mr. Fred R. Dacimo
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT 2 - NRC INTEGRATED INSPECTION REPORT NO. 05000247/2005004

Dear Mr. Dacimo:

On September 30, 2005 the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Indian Point Nuclear Generating Unit 2 (IP2). The enclosed integrated inspection report documents an inspection finding, which was discussed on October 19, 2005, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of the inspection, one finding of very low safety significance (Green) was identified. This finding was determined to be a violation of NRC requirements. However, because of the very low safety significance, and because it was entered into your corrective action program, the NRC is treating the finding as a non-cited violation (NCV) consistent with Section VI.A. of the NRC Enforcement Policy. If you contest the NCV 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, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Senior Resident Inspector at Indian Point 2.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document

Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/ Donald Jackson signing for

Brian J. McDermott, Chief
Projects Branch 2
Division of Reactor Projects

Docket No. 50-247
License No. DPR-26

Enclosure: Inspection Report No. 05000247/2005004
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	05000247/2005004
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point Nuclear Generating Unit 2
Location:	295 Broadway, Suite 3 Buchanan, NY 10511-0308
Dates:	July 1, 2005 - September 30, 2005
Inspectors:	M. Cox, Senior Resident Inspector, IP2 P. Habighorst, Senior Resident Inspector, IP2 C. Long, Acting Resident Inspector, IP2 D. Jackson, Senior Project Engineer, Region 1 T. Hipschman, Senior Resident Inspector, IP3 B. Wittick, Resident Inspector, IP3 G. Bowman, Reactor Engineer, Region 1 J. Noggle, Senior Health Physicist
Approved by:	Brian J. McDermott, Chief Projects Branch 2 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000247/2005004; 07/01/2005 - 09/30/2005; Indian Point Nuclear Generating Unit 2; Operability Evaluations

The report covers a 3-month period of inspection by resident inspectors and three regional inspectors. One Green Non-cited Violation (NCV) was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management reviews. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspector identified a Green NCV for the licensee's failure to properly implement a design modification involving the Safety Injection (SI) pump discharge relief valve, SI-855. This was determined to be a violation of 10CFR50 Appendix B, Part III, Design Control.

  The deficiency was more than minor because it affected the design control attribute of the Mitigating Systems cornerstone objective to ensure availability, reliability and capability of the SI system to prevent undesirable conditions. The issue was a design deficiency that did not result in loss of function per GL 91-18 (rev 1), and was determined to be of very low safety significance (Green) since revised calculations demonstrated the system piping remained capable of performing its specified function. (Section 1R15)

B. Licensee-Identified Violations

None.

Report Details

Summary of Plant Status

Indian Point Unit 2 (IP2) began the inspection period at full power and remained at full power throughout the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope (71111.04 - 3 samples)

Partial System Walkdowns: The inspectors performed 3 partial system walkdowns during periods of system train unavailability in order to verify that the alignment of the available train was proper to support its required safety functions, and to assure that Entergy had identified and properly addressed equipment discrepancies that could potentially impair the capability of the available train. Referenced documents are listed in the Supplemental Information attachment at the end of this report. The following system walkdowns were performed:

- On July 14, 2005, the inspector performed a partial system walkdown of 21 emergency diesel generator (EDG) fuel oil system while the 23 EDG was out of service for maintenance. The inspector reviewed system drawings and the applicable checkoff list to verify proper alignment of valves and control switches. The inspector also observed the physical condition of the equipment during the verification.

- On August 23, 2005, the inspector performed a partial system walkdown of the 21 125VDC bus. The inspector reviewed system drawings and the applicable checkoff list to verify proper alignment of circuit breakers and control switches. The inspector also observed the physical condition of the equipment during the verification.

- On August 24, 2005, the inspectors performed a walkdown of the auxiliary feedwater system. The inspectors walked down the systems using COL 21.3, “Steam Generator Water Level and Auxiliary Boiler Feedwater,” Rev. 27, and the system flow diagram. The inspectors verified that all accessible components were in the proper position per the COL and verified that any position discrepancies were properly documented. The inspectors also verified that the field configuration was consistent with the current revision of the COL. Additionally, the inspectors evaluated the physical condition of the equipment during the walkdown and reviewed open condition reports (CR) and work orders to evaluate if any would potentially impact system performance.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

1. Quarterly Inspection

a. Inspection Scope (71111.05Q - 7 samples)

The inspector toured areas that were identified as important to plant safety and risk significance. The inspector consulted the Indian Point 2 Individual Plant Examination for External Events (IPEEE), Section 4.0, "Internal Fires Analysis," and the top risk significant fire zones in Table 4.6-2, "Summary of Core Damage Frequency Contributions from Fire Zones." The objective of this inspection was to determine if Entergy had adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, and had adequately established compensatory measures for degraded fire protection equipment. The inspector evaluated conditions related to: 1) control of transient combustibles and ignition sources; 2) the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and 3) the fire barriers used to prevent fire damage or fire propagation. Reference material used by the inspector to determine the acceptability of the observed conditions in the fire zones are referenced in the Supplemental Information attachment at the end of this report. The areas reviewed were:

- Zone10, Emergency Diesel Generator building
- Zone 55, Auxiliary Feedwater Pump Room
- Zone 21, 23, Charging Pump Cubicles
- Zone 272, Uninteruptable Power Supply to Proteus and SAS
- Zones 90, 91A, Spent Fuel Pool equipment area
- Zone 4, 21 RHR pump cubicle
- Zones 55, 55A, Transformer Yard

b. Findings

No findings of significance were identified.

2. Annual Inspection

i. Inspection Scope (71111.05A - 1 sample)

On September 21, 2005, the inspectors observed an unannounced fire brigade drill. The drill was conducted in accordance with the licensee's preplanned drill scenario and simulated an electrical fire in the primary auxiliary building. The drill was a routine training exercise for current fire brigade members. The inspectors evaluated the readiness of the fire brigade to suppress and contain the fire, and evaluated the following aspects of the drill:

- The fire brigade properly donned protective clothing/turnout gear.
- Self-contained breather apparatus (SCBA) equipment was properly worn and used.
- Fire hose lines were capable of reaching all necessary fire hazard locations, were laid out without flow restrictions, and were simulated as charged with water.
- Brigade members entered the fire area in a controlled manner.
- Sufficient fire fighting equipment was brought to the scene by the fire brigade.

- The fire brigade leader's fire fighting directions were thorough, clear and effective.
- Radio communications with the plant operators and between fire brigade members were efficient and effective.
- Members of the fire brigade checked for fire victims and propagation into other plant areas.
- Effective smoke removal operations were simulated.
- The fire fighting pre-plan strategies were utilized.
- The licensee's pre-planned drill scenario was followed.
- The drill objectives and acceptance criteria were met.

The inspectors also observed the post-drill critique and evaluated it for thoroughness and degree of critical self-assessment.

b. Findings

No findings of significance were identified.

1R11 Operator Requalification Inspection

a. Inspection Scope (71111.11Q - 1 sample)

On August 8, 2005, the inspectors observed simulator training for licensed operators on Operations Team 2D. The inspectors reviewed an "as found" simulator exercise performed in accordance with lesson plan #SES-E-3, "SGTL, PT-412A Instrument Failure, SGTR". The inspectors reviewed the scenario to ensure it had clear enabling objectives, well defined plant conditions and evaluated the critical task to ensure proper performance by the crew.

During the simulator exercise, the inspectors evaluated the team's performance for: 1) clarity and formality of communications; 2) correct implementation and use of emergency operating procedures (EOP's) and abnormal operating procedures; 3) operators' ability to properly interpret and verify alarms; 4) operators' ability to classify events in a timely fashion, and 5) operators' ability to take timely actions in a safe direction based on transient conditions. In addition, the inspectors evaluated the Control Room Supervisor's ability to exercise effective oversight and control of the crew's actions during the exercise. The inspectors verified that the feedback from the instructors was thorough, that they identified specific areas for improvement, and that they reinforced management expectations regarding crew competencies in the areas of procedure use, communications and peer checking. The inspectors also evaluated Entergy's post-scenario critique to ensure all issues were appropriately identified.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope (71111.12Q - 1 sample)

The inspectors reviewed the maintenance activities listed below, and recent performance issues with systems and components to assess the effectiveness of Entergy's Maintenance Rule (MR) program. Using 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," the inspectors verified that Entergy was implementing their MR program in accordance with NRC regulations and guidelines, properly classifying equipment failures, and using the appropriate performance criteria for MR systems.

The inspectors also reviewed work orders (WOs), and associated post-maintenance test activities to assess whether: 1) the effect of maintenance work in the plant had been adequately addressed by control room personnel; 2) work planning was adequate for the maintenance performed; 3) the acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing documents; and, 4) the equipment was effectively returned to service. Referenced documents are listed in the Supplemental Information attachment at the end of this report. The below-listed systems maintenance activities were observed and/or evaluated.

Nuclear Instrumentation System

The inspectors performed a review of the Nuclear Instrumentation System (NIS) for proper accounting by Entergy for planned and unplanned unavailability of components of the NIS for proper MR tracking. The inspector reviewed several maintenance activities for appropriate MR tracking. The inspector reviewed the NIS MR basis document for appropriate system boundaries between the NIS and other systems and appropriate performance goals. The inspector reviewed the activities involved with the NI-43 current indicator and the installation of temporary alteration 04-1-031 for proper MR evaluation.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope (71111.13 - 5 samples)

The inspectors observed selected portions of emergent maintenance work activities to assess Entergy's risk management in accordance with 10 CFR 50.65(a)(4). The inspectors verified that Entergy took the necessary steps to plan and control emergent work activities, to minimize the probability of initiating events, and to maintain the functional capability of mitigating systems. The inspectors observed and/or discussed risk management actions with maintenance and operations personnel. The following emergent work activities were observed:

- WO IP2-04-10290, 24 DC Bus Ground Troubleshoot and Repair

- CR IP2-2005-02981, Leakage from PCV-455A
- WO IP2-05-00143, 22 MBFP Loss of Trip Circuit
- WO IP2-05-22540, 22 Main Feedwater Regulating Valve (FCV-427) Positioner repair due to erratic operation
- WO IP2-00519, Leak repair downstream of Service Water valve 840

b. Findings

No findings of significance were identified.

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

a. Inspection Scope (71111.14 - 1 sample)

For the non-routine evolution described below, the inspectors reviewed operator logs, plant computer data, strip charts, and observed operations to determine if the activity was performed in accordance with plant procedures.

- The inspectors observed activities associated with Unit 1 fuel cleaning and inspection. The inspectors reviewed Unit 1 technical specifications associated with fuel storage and movement and verified that fuel movement would be performed under the direct supervision of licensed Unit 2 operations personnel as required. The inspectors attended an infrequently performed test and evolution brief to ensure that workers properly understood the associated procedure and the prescribed precautions and limitations. The inspectors reviewed procedure MRS-SSP-1871-IP1, "Indian Point Unit 1 Fuel Handling Procedure", and observed operations on the load floor in preparation for fuel movement including the initial lift of a dummy fuel assembly and basket.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15 - 5 samples)

The inspectors selected operability evaluations that Entergy had generated that warranted review on the basis of potential risk significance. The selected samples are addressed in the CRs listed below. The inspectors assessed the accuracy of the evaluations, the use and control of compensatory measures, if needed, and compliance with the TSs. The inspectors' review included a verification that the operability evaluations were made as specified by procedure ENN-OP-104, "Operability Determinations." The technical adequacy of the evaluations was reviewed and compared to the TS's, Technical Requirements Manual (TRM), FSAR, and associated design basis documents.

- CR IP2-2005-3029, Failure of containment isolation valve SWN-71-4A (24 Fan Cooler Unit service water outlet) to close

- CR IP2-2005-03288, RPS cable separation associated with reactor coolant breaker auxiliary contacts
- CR IP2-2005-03340, Service Water leak on 21 EDG lube oil cooler
- CR IP2-2005-3557, Spent Fuel Pool shrinkage crack and seepage on south wall
- CR IP2-2005-03469, Safety Injection (SI) system relief valve setting

b. Findings

Introduction. The inspector identified a Green NCV due to the licensee's failure to properly implement a design modification involving the SI pump discharge relief valve, SI-855. This was determined to be a violation of 10CFR50 Appendix B, Part III, Design Control.

Description. Due to problems associated with premature lifting and failure of the SI pump discharge header relief valve, SI-855, during system operations and testing, the licensee developed a permanent modification package, MMS-90-04501-M, in April 1991 that involved raising the relief valve setpoint from 1575 psig to 1670 psig.

The licensee developed a modification package MMS-S65-001-0, and supporting calculations of piping maximum allowed pressure based on minimum wall thickness per the guidance of ASME/ANSI USA Standard Code For Pressure Piping B31.1, 1967. The licensing basis for piping per the Safety Analysis Report is ASME/ANSI American Standard Code For Pressure Piping B31.1-1955. The calculations provided a maximum allowed piping pressure of 2242 psi. After determining maximum allowed pressures for flanges (2100 psig) and flow capacity the licensee determined it was acceptable to raise the SI-855 setpoint to 1670 psig.

The Final Safety Analysis Report (FSAR), section 6.2.2.3.6, specifies that the relief valve is to be set at the design pressure of the SI piping. The design pressure specified by Westinghouse is 1575 psig at 200 degrees F. The FSAR guidance is in concurrence with the guidance of ASME/ANSI USA Standard Code For Pressure Piping B31.1, 1967 and ASME Section III, NB-7300/7400, which require that the maximum sustained system pressure is less than the design pressure. In the 10CFR50.59 Safety Evaluation, No. 91-012-MD, for the modification, it was incorrectly determined that there were no changes required to the safety analysis report.

Additionally, the calculations for maximum allowed pressure included an incorrect stress value from the stress tables in ASME/ANSI USA Standard Code For Pressure Piping B31.1, 1967. The stress value used was for a maximum system temperature of 100 degrees F, rather than the 200 degrees F design temperature specified in the FSAR.

Analysis. The inspector determined that the finding is a performance deficiency since the licensee failed to properly implement a plant design change such that the SI-855 relief valve setpoint was raised to greater than design pressure, contrary to the guidance of the FSAR, ASME/ANSI USA Standard Code For Pressure Piping B31.1, 1967 and ASME Section III. The deficiency was more than minor because it affected the design control attribute of the Mitigating Systems cornerstone objective to ensure availability, reliability and capability of the Safety Injection System to prevent undesirable conditions. The issue was a design deficiency that did not result in loss of function per GL 91-18 (rev 1), and was determined to be of very low safety significance (Green) since revised calculations demonstrate the safety-related piping remains capable of performing its specified function.

Enforcement. 10CFR50, Appendix B, Section III, Design Control, states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis for those structures, systems and components are correctly translated into specifications, drawings, procedures and instructions. Contrary to the above, relief valve SI-855 was set above the SI system design pressure and it was identified that calculations used to establish maximum allowed pressure for the piping system utilized incorrect stress values. Because this finding is of very low safety significance and has been entered into the CAP (CR-IP2-2005-03469), this violation is being treated as an NCV, consistent with Section V1.A of the Enforcement Policy: **NCV 05000247/2005004-01, Incorrect setting of relief valve SI-855 above system design pressure and failure to submit required changes to the Safety Analysis Report.**

1R16 Operator Workarounds

a. Inspection Scope (71111.16 - 1 sample)

The inspectors performed a review of operator workarounds and burdens to assess the cumulative effects on system reliability, availability, and the potential for misoperation of a system. The inspectors also toured various areas of the plant to evaluate deficient conditions and their potential impact on operators during EOP and AOP usage. This review included the operator work-around and burden list on August 24, 2005 and the control room deficiencies list. The inspectors reviewed the work control and condition reporting programs to assess the open work request tags and CRs for potential operator workaround consideration. In addition, the inspectors focused on the operator workaround associated with one pressurizer spray valve (PCV-455A) being isolated. The Operational Decision Making Process was followed for this issue, and appropriate compensatory actions were taken. The inspectors used OAP-45, "Operator Burden Program," Rev. 0 and EN-OP-111, "Operational Decision Making Issue Process" to evaluate plant deficiencies and their effects on plant operation.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope (71111.19 - 3 samples)

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

The selected testing activities involved components that were risk significant as identified in the IP2 Individual Plant Examination (IPE). The regulatory references for the inspection included TSs and 10 CFR 50, Appendix B, Criterion XIV, "Inspection, Test, and Operating Status." The following testing activities were evaluated:

- WO IP2-04-26969, Post-maintenance testing on PCV-1310B (auxiliary feed pump steam supply pressure control valve) following 12 year preventive maintenance
- WO IP2-02-65725, Post maintenance testing following replacement of FT-412B (S/G flow transmitter)
- WO IP2-04-26650, Post maintenance testing following inspection and cleaning of 22 Emergency Diesel Generator lube oil and jacket water heat exchanger and replacement of pre lube oil pump

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22 - 4 samples)

The inspectors reviewed surveillance test procedures and observed testing activities to assess whether: 1) the test preconditioned the component tested; 2) the effect of the testing was adequately addressed in the control room; 3) the acceptance criteria demonstrated operational readiness consistent with design calculations and licensing documents; 4) the test equipment range and accuracy were adequate and the equipment was properly calibrated; 5) the test was performed per the procedure; 6) test equipment was removed following testing; and 7) test discrepancies were appropriately evaluated. The surveillance tests observed were based upon risk significant components as identified in the IP2 IPE. The regulatory requirements that provided the acceptance criteria for this review were 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," Criterion XIV, "Inspection, Test, and Operating Status," Criterion XI, "Test Control," and TS 6.8.1.a. The following test activities were reviewed:

- 2-SOP-1.7, "Reactor Coolant System Leakage", Rev.40
- PT-M48, "480 Volt Undervoltage Alarm", Rev. 8
- 2PT-Q89, "Control Rod Exercise", Rev. 2
- PT-Q59, "Containment Pressure Bistables", Rev. 8

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23 - 1 sample)

The inspectors reviewed a temporary modification to the plant to ensure that the effects on plant operation were well understood and to ensure that no unintended adverse consequences would result from the modification. The inspector evaluated the modification documentation for accuracy and completeness, the basis for the modification and any associated procedures / changes to procedures to control the temporary modification operation. The following temporary modification was reviewed.

- TA-05-2-094, Defeat 22 MBFP Thrust Bearing Wear Alarm

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope (71114.06 - 1 sample)

The inspectors observed an Emergency Preparedness drill conducted on September 21, 2005. The inspectors used NRC Inspection Procedure 71114.06, "Drill Evaluation" as guidance and criteria for evaluation of the drill. The drill consisted of a fire in the Primary Auxiliary Building and resulting in a loss of Component Cooling Water containment isolation valves, a subsequent Reactor Coolant Pump (RCP) thermal barrier leak and a fuel clad failure. The inspectors observed the drill and conducted reviews from the participating facilities onsite, including the IP3 Plant Simulator, the Technical Support Center (TSC), and the Emergency Operations Facility (EOF). The inspectors focused the reviews on the identification of weaknesses and deficiencies in the classification and notification timeliness and quality and accountability of essential personnel during the drill. The inspectors were briefed on Entergy's critique results and compared the NRC-identified weaknesses and deficiencies to those identified by Entergy to ensure that problem areas were properly identified.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope (71121.01 - 8 samples)

On July 11-15, 2005, the inspectors conducted the following activities during normal plant operating conditions to verify that the licensee was properly implementing physical, engineering, and administrative controls for access to high radiation areas, and other radiologically controlled areas, and that workers were adhering to these controls when working in these areas. Implementation of the access control program was reviewed against the criteria contained in 10 CFR 20, site technical specifications, and the licensee's procedures.

(1) The following exposure significant work area was evaluated to determine if radiological controls (e.g., surveys, postings, and barricades) were acceptable.

- CVCS pump pit

(2) The radiation work permit (RWP) associated with the above work activity was reviewed with respect to high radiation area controls including electronic dosimeter alarm set points.

(3) With respect to the work activity listed in (1) above, a walk down of this work area was conducted with a radiation survey instrument to determine whether radiation work permit (RWP), procedure, and engineering controls were in place, and whether licensee surveys and postings were complete and accurate, and that air samplers were properly located.

(4) The work activity listed in (1) above was reviewed against the radiological control requirements as specified in the applicable RWP and ALARA review, as well as verbal instructions provided by radiation protection (RP) technicians during radiological briefings to workers.

(5) With respect to the work activity listed in (1) above, the conduct of necessary system breach survey and evolving radiological hazards associated with work activities were observed to evaluate the radiation protection job coverage and contamination controls.

(6) During observations of the work activity listed in (1) above, radiation worker performance was evaluated with respect to radiological work requirements and radiological briefing instructions.

(7) The inspectors toured the accessible areas of Units 1, 2, and 3 and verified the adequacy of radiological postings and verified the locking of all high dose rate high radiation areas and very high radiation areas as required.

(8) There were no licensee internal dose assessments greater than 50 mrem CEDE during 2005 at Indian Point Energy Center.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope (71121.02 - 6 samples)

On July 11 - 15, 2005, the inspector conducted the following activities to verify that the licensee was properly maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR 20.1101(b) and the licensee's procedures.

(1) The plant collective exposure history trend and current three-year rolling average collective exposure data was reviewed. Based on 2002-2004 exposure data, Indian Point Unit 2 performance of 152 person-rem ranks in the fourth quartile, and Indian Point Unit 3 performance of 36 person-rem ranks in the first quartile of U.S. pressurized water reactors.

(2) The following highest exposure work activities for the Unit 3 Spring 2005 refueling outage were selected for review.

- reactor coolant pump work
- outage valve work
- reactor disassembly / reassembly
- scaffold building and inspection
- radiation protection support

(3) The ALARA reviews for the outage work activities listed in (2) above were evaluated with respect to initial exposure estimates and any subsequent credits due to emergent work or increased dose rates, and then compared to the actual exposure results obtained. Any causes for exposure overruns were identified and quantified where appropriate.

(4) With respect to the ALARA reviews that were evaluated in (3) above, the methods for adjusting exposure estimates were reviewed relative to changes in work scope or increased dose rates in order to preserve the original work activity exposure performance measurement of the work activities.

(5) The site specific trend in source term was reviewed and found to be stable at approximately 70 mrem/hr average intermediate loop piping for Unit 2 and a decreasing trend at approximately 20 mrem/hr for Unit 3. This compares favorably with the industry average of 100 mrem/hr.

(6) The following licensee self-assessments and audits related to the ALARA program were reviewed to determine if the licensee's overall audit program scope and frequency met the requirements of 10 CFR 20.1101.

- Radiation Protection Department Annual Self-Assessment Report, June 2004 - June 2005
- TID-04-008, Evaluation for the Temporary Storage of Radioactive Materials within the Protected Area, June 30, 2005
- TID-05-002, Prospective Evaluation of the Need for Internal Monitoring for Radiation Workers, June 30, 2005

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Materials Processing and Shipping

a. Inspection Scope (71122.02 - 6 samples)

During the period September 19-23, 2005, the inspector conducted the following activities to verify that the licensee's radioactive material processing and transportation programs complied with the requirements of 10 CFR 20, 61, and 71; and Department of Transportation (DOT) regulations 49 CFR 170-189.

(1) The inspector reviewed the solid radioactive waste system description in Chapter 11 of the final safety analysis reports (FSAR) for Units 2 and 3, the 2003 radiological effluent release reports for Units 2 and 3 for information on the types and amounts of radioactive waste disposed, and the scope of the licensee's audit program to verify that it meets the requirements of 10 CFR 20.1101.

(2) The inspector walked-down the liquid and solid radioactive waste processing systems to verify and assess that the current system configuration and operation agree with the descriptions contained in the FSAR and in the Process Control Program (PCP); and reviewed the status of any radioactive waste process equipment that is not operational and/or is abandoned in place; verified that the changes were reviewed and documented in accordance with 10 CFR 50.59, as appropriate.

(3) The inspector reviewed the radio-chemical sample analysis results for each of the licensee's radioactive waste streams (primary resin Unit 2/ Unit 3, liquid waste system resin Unit 2/ Unit 3, dry active waste Unit 2/ Unit 3, and Unit 1 east spent fuel pool sludge); reviewed the licensee's use of scaling factors and calculations with respect to these radioactive waste streams to account for difficult-to-measure radionuclides; verified that the licensee's program assures compliance with 10 CFR 61.55 and 10 CFR 61.56 as required by Appendix G of 10 CFR Part 20; and, reviewed the licensee's program to ensure that the waste stream composition data accounts for changing operational parameters and thus remains valid between the annual or biennial sample analysis update.

(4) The inspector observed shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifests, shipping papers provided to the driver, and licensee verification of shipment readiness; verified that the receiving licensee is authorized to receive the shipment packages; and, observed radiation workers during the preparation and shipment of shipment no. 05-176 on September 21, 2005 to Duratek, Oak Ridge, TN. The inspector determined that the shipper was knowledgeable of the shipping regulations and that shipping personnel demonstrate adequate skills to accomplish the package preparation requirements for public transport with respect to NRC Bulletin 79-19 and 49 CFR Part 172 Subpart H, and verified that the licensee's training program provides training to personnel responsible for the conduct of radioactive waste processing and radioactive material shipment preparation activities.

(5) The inspector sampled the following non-excepted package shipment records and reviewed these records for compliance with NRC and DOT requirements.

- 05-176, Unit 2 and 3 DAW shipment to Duratek on September 21, 2005
- 05-165, Unit 2 bead resin shipment to Studsvik on September 8, 2005
- 05-046, Unit 2 reactor coolant pump motor shipped to Curtis Wright EMD on February 25, 2005
- 04-117, Unit 3 primary resin shipped to Studsvik on August 20, 2004
- 04-100, Unit 2 and 3 DAW shipment to Duratek on August 3, 2004
- 04-053, Unit 3 bead resin shipment to Studsvik on April 14, 2004
- 04-010, Unit 3 reactor vessel capsule shipment to Westinghouse on January 16, 2004

(6) The inspector reviewed the licensee's Licensee Event Reports, Special Reports, audits, State agency reports, and self-assessments related to the radioactive material and transportation programs performed since the last inspection and determined that identified problems are entered into the corrective action program for resolution. The inspector also reviewed corrective action reports written against the radioactive material and shipping programs since the previous inspection.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

1. Daily Review

a. Inspection Scope (71152)

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive failures or specific human performance issues for follow-up, the inspectors screened all items entered into Entergy's corrective action program (CAP). This review was accomplished by reviewing copies of each condition report (CR).

b. Findings

No findings of significance were identified.

2. Occupational Radiation Safety Cornerstone

a. Inspection Scope (71121)

The inspector reviewed 10 corrective action CRs that were initiated between April 2005 and June 2005, and were associated with the radiation protection program. The inspector verified that problems identified by these CRss were properly characterized in the licensee's event reporting system, and that applicable causes and corrective actions were identified commensurate with the safety significance of the radiological occurrences.

b. Findings

No findings of significance were identified.

3. Public Radiation Safety Cornerstone

a. Inspection Scope (7112201)

The inspector reviewed nine corrective action CRs that were initiated between January 2004 and August 2005 and were associated with the radwaste transportation program. The inspector verified that problems identified by these CRs were properly characterized in the licensee's event reporting system, and that applicable causes and corrective actions were identified commensurate with the safety significance of the radiological occurrences.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

TI 2515/161 - Transportation of Reactor Control Rod Drives (CRD) in Type A Packages

a. Inspection Scope

This area was inspected to verify that the licensee's radioactive material transportation program complies with specific requirements of 10 CFR Parts 20, 71, and Department of Transportation regulations contained in 49 CFR Part 173. The inspector interviewed licensee personnel and determined the licensee had undergone refueling/defueling activities between January 1, 2002, and present, but it had not shipped irradiated CRDs drives in Department of Transportation Specification 7A Type A packages.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Exit Meeting Summary

On October 19, 2005, the inspectors presented the inspection results to you and other Entergy staff members, who acknowledged the inspection results presented. The inspectors asked the licensee what materials examined during the inspection should be considered proprietary. No proprietary information is presented in this report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

E. Anderson	Lead Engineer, Cable Separation Program Improvements
V. Andreozzi	Systems Engineering Electrical Systems Supervisor
T. Barry	Security Manager
T. Beasley	System Engineer
L. Bello	Records Management Clerk
T. Foley	System Engineer
C. Wend	Radiation Protection Manager
C. Bergeren	In-Service Testing Engineer
M. Sicard	I&C Superintendent
J. Comiotes	Director, Nuclear Safety Assurance
P. Conroy	Manager, Licensing
F. Dacimo	Site Vice President
G. Dahl	Technical Specialist, Licensing
G. Dean	Assistant Operations Manager - Training
R. DeCensi	Technical Support Manager
R. Drake	Supervisor, Mechanical Design Engineering
D. Gately	Assistant Radiation Protection Superintendent
G. Hinrichs	Project manager
F. Inzirillo	Emergency Planning Manager
T. Jones	Nuclear Safety/Licensing Specialist, Licensing
L. Lee	Systems Engineering Primary Systems Supervisor
W. Mahlmeister	Technical Lead, Cable Separation Program Improvements
D. Mayer	Unit 1 Project Manager
B. Meek	System Engineer
T. Orlando	Systems Engineering Manager
P. Peloquin	Project Engineer
S. Petrosi	Manager, Design Engineering
J. Raffaele	Supervisor, Electrical Design Engineering
V. Renzi	Contractor, EPM (Software Support and Operations Manager)
B. Rokes	Licensing Engineer
P. Rubin	Plant Manager
H. Santis	Project Construction Manager
C. Schwarz	Vice President, Operations Support
G. Schwartz	ISFSI Project Manager
J. Skonieczny	Project Engineer
A. Stewart	Licensing
D. Smith	Scheduling and Work Order Coordinator
R. Sutton	Systems Engineer
J. Tuohy	Manager, Cable Separation Program Improvements
J. Ventosa	Engineering Manager
A. Vitale	Site Operations Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000247/2005004-001	NCV	Inadequate design controls during SI relief setpoint change resulting in relief pressure exceeding system design pressure (Section 1R15)

LIST OF BASELINE INSPECTIONS PERFORMED

71111.04	Equipment Alignment	1R04
71111.05	Fire Protection	1R05
71111.11	Operator Requalification Inspection	1R11
71111.12	Maintenance Effectiveness	1R12
71111.13	Maintenance Risk Assessment and Emergent Work Control	1R13
71111.14	Personnel Performance During Non-routine Plant Events	1R14
71111.15	Operability Evaluations	1R15
71111.16	Operator Workarounds	1R16
71111.19	Post-Maintenance Testing	1R19
71111.22	Surveillance Testing	1R22
71111.23	Temporary Plant Modifications	1R23
71114.06	EP Drill	1EP6
71121.01	Access Control to Radiologically Significant Areas	2OS1
71121.02	ALARA Planning and Controls	2OS2
71130.08	Fitness For Duty Program	
71153	Event Followup	4OA3

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures
COL 27.3.1, Diesel Generators, revision 23
COL 21.3, Steam Generator Water Level and Auxiliary Boiler Feed Water, revision 23
COL 27.1.6, Instrument Buses, DC Distribution and PA Inverter, revision 21

Condition Reports

IP2-2005-2915
IP2-2005-2918
IP2-2005-2919
IP2-2005-2921
IP2-2005-3073

Drawings

9321-F-2030-38, Flow Diagram Fuel Oil to Diesel Generators
A208501-36, 125 VDC Distribution Panels

9321-F-2019-112, Flow Diagram for Boiler Feedwater

Section 1R05: Fire Protection

Procedures

PFP-263, “Transformer Yard-Exterior Buildings,” Rev. 0
PFP-258, "EDG #21 - #22 - #23 - Diesel Generator Building," Rev. 0
PFP-259, "Auxiliary Feedwater Pump Room - Auxiliary Feedwater Building," Rev. 0
PFP-217, “General Floor Plan - Fuel Storage Building”, Rev. 0
PFP-152, UPS to Proteus and SAS “General Floor Plan - Superheater Building,” Rev. 0
Local Alarm Response Procedure 38-1, “UPS Room Fire Panel Outside UPS Room,” Rev. 2
COL 29.6, “Fire Protection System.” Rev. 52, pp 31, 32
SOP 29.6, “Fire Protection System Operation.” Rev. 22

SMM-DC-901, "IPEC Fire Protection Program Plan," Rev. 2
ENN-DC-161, "Transient Combustible Program," Rev. 1
Fire Protection Implementation Plan- Fire Zone 4- Residual Heat Removal Pump 21 Room, Rev 8.
Fire Protection Implementation Plan- Fire Zone 7- Charging Pump Room Number 23, Rev 4.
PFP-204,"General Floor Plan- Primary Auxiliary Building- 15'-0" El,” Rev. 0
PFP-211,"General Floor Plan- Primary Auxiliary Building- 80'-0" El,” Rev. 0

Condition Reports
IP2-2004-00538
IP2-2003-27107
IP2-2005-03426

Miscellaneous
Work Order IP2-05-22336

Section 1R11: Operator Requalification

Miscellaneous
Lesson Plan SES-E-3

Procedures
AOP-INST-1, “Instrument or Controller Malfunction”
AOP-SG-1, “Steam Generator Tube Leak”
EOP E-0, “Reactor Trip or Safety Injection”
EOP E-3, “Steam Generator Tube Rupture”
IP-EP-120, “Emergency Classification”

Section 1R12: Maintenance Effectiveness

Condition Reports:
IP2-2004-00398
IP2-2005-02045
IP2-2004-06408
IP2-2004-01656

IP2-2004-03375
IP2-2005-01146

Miscellaneous
Work Order IP2-04-28507
Indian Point Energy Center IP2 andIP3 Maintenance Rule Basis Document for Nuclear Instrumentation System, Rev. 0

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

Work Orders
IP2-04-10290
IP2-04-18586
IP2-05-00143
IP2-05-21068
IP2-05-22540
IP2-04-10290
IP2-05-00519
IP2-05-23050
IP2-05-23534
IP2-05-23569
IP2-05-22807

Condition Reports:
IP2-2005-02981
IP2-2005-03120
IP2-2005-03120
IP2-2004-02410
IP2-2005-03494
IP2-2005-03743
IP2-2005-03703
IP2-2005-03496

Miscellaneous

ODMI, “PCV-455A Isolation due to packing leak”
Troubleshooting Plan for RCS Leakage into RCDT
Valve packing Data Sheet - PCV-455A
IP-SMM-103 “Troubleshooting Data Sheet”
EN-MA-125 “Troubleshooting and Repair”
24 Battery Charger Ground Troubleshooting Plan
IP2 Probabalistic Safety Assessment, Appendix C, Internal Flooding Analysis Notebook Rev. 0
UltraSonic Examination Report 05UT219
UltraSonic Examination Report 05UT220
UltraSonic Examination Report 05UT226
IP-CALC-05-00787
IP-CALC-05-00721
Individual Plant Examination of External Events for Indian Point Unit 2, December 1995
2-AOP-Flooding-1, Rev. 1
2-AOP-SW-1, Rev. 3

Drawings
9321-F-2682-13
9321-F-2719-128
9321-LL-3140-31
A209762-66

Section 1R15: Operability Evaluations

Calculations
MMS-565-001-0, RV855 Max Pressure Setting

Condition Reports
IP2-2005-3557
IP2-2005-03288
IP2-2005-3029
IP2-2005-03340
IP2-2004-06167
IP2-2004-04277

Drawings
UFSAR Figure 7.2-23
A225104-06
9321-LL-3114-16
110E073
263989-02
A209762-66
D251232-11

Miscellaneous
Work Order IP2-04-32170
Work Order IP2-04-26969
ENN-DC-171, Screening and Functional Failure Determination Form
2-PT-Q034B, “PCV-1310A and PCV-1310B Nitrogen Supply
ASME/ANSI American Standard Code For Pressure Piping B31.1-1955.
ASME/ANSI USA Standard Code For Pressure Piping B16.5 - 1968, “Steel Pipe Flanges and Flanged Fittings”
ASME/ANSI USA Standard Code For Pressure Piping B31.1, 1967
ASME Article NB-7000, “Overpressure Protection”
Indian Point 2 FSAR, Chapter 6, “Safety Injection System”
WAPD E-Spec G-569866, Rev No. 2, “Piping Specification, Class 1501"
Consolidated Edison Co. Indian Point Station Modification No. MMS-90-04501-M

Section 1R19, Post Maintenance Testing

Condition Reports
IP2-2002-11276

Work Orders

IP2-04-26969
IP2-02-65725
IP2-04-26553
IP2-05-17069
IP2-05-00782
IP2-04-26650
IP2-04-26553
IP2-05-16932
IP2-04-26551
IP2-04-24466
IP2-04-21389
IP2-05-16931

Drawings
9321-F-2019-112
A225338-9
D260516-02

Miscellaneous
2-PT-M021B, “Emergency Diesel Generator 22 Load Test”, Rev. 14
Vendor Manual “Model 1152 Alphaline Nuclear Pressure Transmitter”

Section 1R22: Surveillance Testing

Procedures
2-SOP-1.7, Rev 40, “Reactor Coolant System Leakage”
2-PT-M048, “480 Undervoltage Alarm Test” Rev. 18
2-PT-Q089, Control Rod Drive Exercise” Rev. 2
Pt-Q59, “Containment Pressure Bistabes”, Rev.8

Condition Reports
IP2-2005-3062

Miscellaneous
IP2 RCS Leakrate Calculation, Version 1.1
Pre-job Brief Sheet - PT-M48 480v Undervoltage

Drawings
9321-LL-3117-32

Section 1R23, Temporary Alterations

Procedures
2-ARP-FBF, MBFPT Thrust Bearing Wear

Condition Reports
IP2-2005-03051

Work Orders
IP2-05-00135

Miscellaneous
Temporary Alteration Package, TA-05-2-094

Section 1EP6: Emergency Plan Drill

Procedures
IP-EP-120, Emergency Classification, Rev. 1
IP-EP-210, Central Control Room, Rev. 1
IP-EP-250, Emergency Operations Facility, Rev. 5
3-AOP-SSD-1, Rev 4, Control Room Inaccessibility Safe Shutdown Control
3-ONOP-FP-1, Plant Fires
IP-EP-360, Core Damage Assessment

Condition Reports
IP3-2005-04481
IP3-2005-04482
IP3-2005-04483
IP3-2005-04484

Miscellaneous
September 21, 2005 Training Drill Scenario

Section 20S1: Access Control to Radiologically Significant Areas

Westinghouse Issue Report No. 05-89-M004, June 10, 2005

Procedures

Reactor Coolant Pump Back Seating Procedure, O-PMP-402-RCS, Rev. 1
HRA/LHRA/REA/VHRA Boundary Verifications, RP-STD-17
ALARA Program, IP-SMM, RP-301
RWP Preparation and ALARA Planning, O-RP-RWP-400, Rev. 2

Condition Reports

CR-IP2-2005-1444	CR-IP3-2005-2209	CR-IP3-2005-2219
CR-IP2-2005-1604	CR-IP3-2004-2448	CR-IP3-2005-2584
CR-IP3-2005-2797	CR-IP3-2005-2799	CR-IP2-2005-2679
CR-IP3-2005-1794		

Section 2PS2: Radioactive Materials Processing and Shipping

Condition Reports

IP2-2004-01163
IP2-2004-06209
IP2-2005-00317
IP2-2005-00613
IP2-2005-01441

IP2-2005-01891
IP2-2005-02679

Quality Assurance Audit no. QA-15-2005-IP-1: IPEC Radiological Waste Program, September 2005

NUPIC Audits: Framatome ANP, December 2003; Duratek - Barnwell, April 2003; Barnwell - Oakridge and Kingston, TN, May 2003; Studsvik, October 2004; RACE, January 2003

Procedures

Process Control Program, RW-SQ-4.007, Rev. 9, Solid Radioactive Waste Process Control Program, RE-PCP, Rev. 7

LIST OF ACRONYMS

ALARA	As Low As is Reasonably Achievable
ASME	American Society of Mechanical Engineers
CAP	corrective actions process
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
CR	condition report
CRD	control rod drive
CVCS	chemical and volume control system
DAW	dry active waste
DOT	U. S. Department of Transportation
EDG	emergency diesel generator
EOF	Emergency Operations Facility
EOP	emergency operating procedure
FSAR	final safety analysis report
IMC	inspection manual chapter
IN	Information Notice
IP	Inspection Procedure
IP2	Indian Point 2
IPE	individual plant examination
IPEC	Indian Point Energy Center
IPEEE	individual plant examination of external events
IR	Inspection Report
ISFSI	Indian Point independent spent fuel installation facility
MBFP	main boiler feed pump
MR	maintenance rule
NCV	non cited violation
NRC	Nuclear Regulatory Commission
NIS	nuclear instrumentation system
PCP	process control program
PWT	post work test
RCP	reactor coolant pump
RP	radiation protection
RPS	reactor protection system

RWP	radiation work permit
SAS	secondary alarm station
SCBA	self-contained breathing apparatus
SDP	significance determination process
SGTR	steam generator tube rupture
SI	safety injection
TRM	technical requirements manual
TS	technical specification
TSC	Technical Support Center
WO	work order