Indian Point 2
3Q/2003 Plant Inspection Findings

Initiating Events

Significance: G May 15, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
IMPROPER EMERGENT WORK PACKAGE INSTRUCTIONS FOR 22 STEAM GENERATOR LEVEL BISTABLE REPLACEMENT
On February 7, 2003, a self-revealing finding involved inadequate emergent work instructions that resulted in an electrical short during replacement of the 22 steam generator low level bistable. The electrical short caused a breaker trip on circuit 10 of instrument bus 21and the resultant loss of electrical power to the pressurizer level and reactor coolant system pressure control channels (failed low). The inadequate work instructions is considered a non-cited violation of 10 CFR 50 Appendix B, Criterion V, since the instructions did not account for consideration of performing this replacement with the circuit de-energized or the proximity to other reactor protection system relays.

The performance issue is more than minor since the operators were required to take action to restore reactor coolant system pressure and pressurizer level to preclude a reactor trip. The finding involves the initiating events cornerstone in that it increased the likelihood of upset in plant stability and it involves human error during the planning of an emergent work activity. This finding is considered to be of very low safety significance in that in accordance with NRC Manual Chapter 0609, Appendix A, the finding did not contribute to the likelihood of a secondary or primary LOCA initiator and it did not contribute to either a reactor trip or mitigation system unavailability.
Inspection Report# : 2003003*(pdf)*

Significance: G Feb 03, 2003
Identified By: NRC
Item Type: FIN Finding
A failure to initiate a condition report to identify problems associated with nonsafety related steam generator level controller replacements.
Green. A failure to initiate a condition report to identify problems associated with nonsafety related steam generator level controller replacements on August 13, 2002, which resulted in a steam generator level transient and required operator action to prevent a reactor trip.

This issue is more than minor because the problem could reasonably be viewed as a precursor to a significant event. Further, the controller replacement had an actual impact on feedwater flow and steam generator level control which required operator action to preclude a reactor trip. This issue affects the initiating event cornerstone objective of limiting conditions that affect plant stability. The finding was determined to be of very low safety significance (Green) because, although it affected stability of plant operating parameters, it did not increase the likelihood of a primary or secondary loss of coolant accident (LOCA), did not contribute to a reactor trip and a loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flooding condition.
Inspection Report# : 2003002*(pdf)*

Significance: G Feb 03, 2003
Identified By: NRC
Item Type: FIN Finding
A self-revealing finding was identified for ineffective corrective actions to prevent main feedwater flow and steam generator level transients.
Green. A self-revealing finding was identified for ineffective corrective actions to prevent main feedwater flow and steam generator level transients during installation of a modification to replace nonsafety related steam generator system level controllers.

The corrective actions from problems experienced during controller replacements on August 6 were ineffective to ensure that subsequent controllers replaced on August 9, August 13, and October 7, 2002 did not result in similar steam generator level transients and necessitate operator actions to prevent reactor trips. While this and the previous finding both concern problems with steam generator level replacements, the findings are distinct in that the previous finding identifies problems not entered into the corrective action program, while this finding concerns the ineffectiveness of corrective actions for problems that were entered into the corrective action program. This issue is more than minor because the problem could reasonably be viewed as a precursor to a significant event, since the controller replacements had an actual impact on feedwater flow and steam generator level control which required operator action to preclude a reactor trip. This issue affects the initiating event cornerstone objective to limit conditions that challenge plant stability. However, the finding was similarly determined to be of very low safety significance (Green) because, although it affected stability of some plant parameters, it did not increase the likelihood of a primary or secondary LOCA, did not contribute to a reactor trip and a loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flooding condition

Inspection Report# : 2003002*(pdf)*

Mitigating Systems

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: FIN Finding
THE PERFORMANCE FINDING INVOLVED INADEQUATE SHORT TERM CORRECTIVE ACTIONS ASSOCIATED WITH FIRE LEAKS ON A FIRE HEADER IN THE UNIT 1 TURBINE BUILDING
The inspectors identified a finding involving inadequate corrective actions associated with multiple leaks on a six-inch fire header in the Unit 1 turbine building. On September 10, 2003, an 80 gallon per minute fire header leak occurred that operators isolated by depressurizing the entire fire water suppression system at Unit 2 for approximately three hours. This leak occurred approximately one foot from a similar through-wall leak which occurred on July 16, 2003.

This performance issue is considered more than minor based on example 4.f. in MC 0612 Appendix E. The performance finding involves the Mitigating Systems Cornerstone objective of fire suppression system availability to respond to fires. The finding is very low risk significance based upon the results from the fire protection risk significance screening methodology (FPRSSM). The finding impacts both manual suppression capability and automatic suppression capability.
Inspection Report# : 2003011*(pdf)*

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
NCV OF 10 CFR 50, APP B, CRITERION V. THE PROCEDURAL STEPS FOR THE INSTALLATION OF A FLEXIBLE COUPLING WERE NOT ADEQUATE TO VERIFY THAT THE COMPONENT WAS PROPERLY INSTALLED.
The inspector identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V. In November 2002, a maintenance work instruction to install a 21 emergency diesel generator (EDG) service water supply flexible coupling did not include critical installation steps per the vendor manual. This resulted in a significant service water leak from the expansion joint on August 14, 2003.

This finding is greater than minor since if left uncorrected, it could be a more significant safety concern as this type of flexible coupling is used on all three EDGs. The inspectors determined that the expansion joint leakage was of a very low safety significance since it did not adversely impact service water cooling to the emergency diesel generator or the overall service water system cooling capability, did not impact equipment and functions associated with internal flooding in the diesel generator room, and did not result in a loss of service water or emergency power safety function that contributed to internal flooding initiated events.
Inspection Report# : 2003011*(pdf)*

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
NCV OF 10 CFR 50, APP B. A DESIGN CHANGE PACKAGE DID NOT ACCURATELY REFLECT ACTUAL PLANT CONDITIONS AND RESULTED IN AN UNINTENDED PLANT TRANSIENT
The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, involving the design change package (DCP-200105716-I) to replace a pressurizer level recorder which did not contain accurate design details. As a consequence, during installation of the design change an unintended plant transient challenged operators.

This finding is greater than minor based upon NRC Manual Chapter 0612, Appendix E, example 4.b. This finding is of very low safety significance. The finding did contribute to the likelihood of a reactor trip; however, it did not impact the availability of mitigation equipment, increase the likelihood of a primary or secondary system LOCA, or increase the likelihood of an internal fire or flood.
Inspection Report# : 2003011*(pdf)*

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
EQUIPMENT TAGOUT TO RESTORE THE 22 SEAL INJECTION FILTER WERE INADEQUATE TO MAINTAIN PROPER CONFIGURATION CONTROL OF THE SYSTEM
The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, involving an incomplete procedure for restoring to service the 22 seal injection filter from maintenance. The consequence was an approximate 70 gallon per minute chemical volume and control system leak through an open vent valve which lasted for approximately two minutes before operators identified and shut the vent valve.

This finding is more than minor since it adversely impacted the Mitigating System Cornerstone objective of safety system capability and availability with respect to the attributes of configuration control and procedural quality. The inadequate restoration procedure resulted in a significant chemical and volume control system leak (the capacity of one

coolant charging pump) that degraded normal charging flow and emergency boration capability for a short period of time. The finding is of very low safety significance since it did not result in a loss of emergency boration safety function.
Inspection Report# : 2003011*(pdf)*

Significance: G Jun 28, 2003
Identified By: NRC
Item Type: FIN Finding
INADEQUATE OPERABILITY EVALUATION FOR THE 13.8 KV SYSTEM
The inspector identified that the licensee's operability evaluation during a 13.8 KV system reduced voltage test was inadequate. The operability evaluation did not evaluate accident load carrying capability as defined in the technical specification basis and it did not address communications and protocols between the distribution company and the licensee to restore from the test in a timely manner. NRC Manual Chapter 9900 states that when a system's capability is degraded to a point where it cannot perform with reasonable assurance of reliability, the system should be judged inoperable.

The finding was more than minor because it impacted the attribute of the mitigating system cornerstone objective. Specifically, the cornerstone objective is to ensure that the 13.8 KV system is capable of performing its safety function during a postulated loss of normal power event without undesirable consequences. This finding was determined to be of low safety significance because it did not result in the actual loss of the offsite power supply safety function.
Inspection Report# : 2003007*(pdf)*

Significance: G Jun 28, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Ineffective corrective actions associated with an unauthorized modification to the No. 22 component cooling water pump.
The inspector identified a non-cited violation of 10 CFR 50 Appendix B, Criterion XVI. The licensee did not evaluate and take effective corrective actions associated with a material substitution for the 22 component cooling water pump inboard bearing oil level indication system. The bearing oil level indication system contributed to the failure of the #22 CCW pump on December 5, 2002.

This finding is greater than minor since it is associated with the design control attribute of the mitigating systems cornerstone and affected the cornerstone objective. The inspectors conducted a Phase 1 SDP screening and determined that the failure to take effective corrective action on #22 CCW pump was of a very low safety significance since the redundant train components were operable and unaffected by this unauthorized modification.
Inspection Report# : 2003007*(pdf)*

Significance: G May 15, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
INEFFECTIVE CORRECTIVE ACTIONS ASSOCIATED WITH THE 23 EDG LOAD SWINGS BETWEEN MAY 2000 AND FEBRUARY 2003
The inspectors identified that ineffective corrective actions resulted in repetitive surveillance test failures of the 23 emergency diesel generator between December 2001 and February 2003. This finding is considered a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI. The finding is more than minor because the surveillance test failures impacted the availability of one train of emergency AC power source. This finding was of very low risk

significance because the repetitive failures did not result in an actual loss of function for the emergency AC power.
Inspection Report# : 2003003*(pdf)*

Significance: G May 15, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
POST-WORK TEST INADEQUAATE FOR 22 BORIC ACID TRANSFER PUMP BORIC ACID FILTER STOP VALVE
A self-revealing event was identified on February 26, 2003, when operators observed no boric acid flow to the reactor vessel via the No. 22 boric acid transfer pump (BATP). It was determined that during preventative maintenance activities in March 2001, the post-work test on the No. 22 BATP outlet valve to the boric acid filter stop was inadequate to identify that the valve finger plate was installed upside down. This finding is considered a non-cited violation of 10 CFR 50 Appendix B, Criterion V. This event is considered more than minor because the improperly installed valve plate affected the availability of one train of emergency boration. This is considered to be of very low risk significance in accordance with NRC MC 0609 Appendix A, since the emergency boration function was not lost due to this performance issue.
Inspection Report# : 2003003*(pdf)*

Significance: G Dec 28, 2002
Identified By: NRC
Item Type: FIN Finding
UNTIMELY OPERABILITY DETERMINATION FOR THE 21 DIESEL GENERATOR
On October 9, 2002, the licensee's organization did not identify in a timely manner that the 21 emergency diesel generator was inoperable. The causes for the untimely operability evaluation were fragmented communications between Entergy departments, untimely drip tank sample results, system engineering turnover, and a lack of sensitivity to a loss of the emergency power source safety function. The time between when the non-licensed operator had reported and added inventory to the jacket water expansion tank to the time the emergency diesel generator was declared inoperable was 7.5 hours which exceeded the limiting condition for operation within TS 3.0.1 to be in hot shutdown within seven hours. In the absence of reasonable expectation that a component is operable, the component shall be declared inoperable immediately. The untimely operability evaluation affects the mitigating systems cornerstone objective. The attribute is human performance pre-event. This finding is of very low safety significance in phase 1 of the SDP since the 21 EDG was subsequently declared inoperable and actions within the TS were adhered to. This finding did not result in an actual loss of the emergency on-site power source safety function nor did it increase the risk significance for external events. No violations of NRC requirements were identified.
Inspection Report# : 2002007*(pdf)*

Significance: G Dec 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO IDENTIFY THE CAUSE OF 23 EDG OUTPUT BREAKER TO CLOSE
The inspector identified that Entergy did not adequately evaluate the cause blown control power fuses on the 23 EDG output breaker cubicle on November 10, 2002, that subsequently caused the breaker from closure on November 14, 2002. On November 14, 2002, Entergy identified the cause of the breaker failure as improper operation of the inertial latch mechanism. This performance issue is being treated as a Non-cited Violation of 10 CFR 50 Appendix B, Criterion XVI. This violation is more than minor because the failure to identify the cause and preclude recurrence was considered a precursor to a more significant safety issue, in that, the plant could have started up with only 2 available EDGs - a violation of TS - and not have know it for approximately one month. The issue was determined to be of very

low safety significance (Green) in accordance with MC 0609 Appendix G, since greater than three offsite and onsite power sources were available to cope with a postulated loss of offsite power.
Inspection Report# : 2002007*(pdf)*

Significance: Dec 28, 2002
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
INADEQUATE POST WORK TEST ON STEAM STOP CHECK VALVE
The post work test on the 22 steam generator stop check valve (MS-41) failed to identify that the valve plug was installed upside down. This self-revealing event was identified on November 20, 2002, when operators responded to steam leak-by from this tagged closed valve that resulted in a fire alarm in the auxiliary feedwater pump room. This finding is considered a Non-Cited Violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," in that, the post work test, PT-R67A, Reverse Flow Check at MS-41 and MS-42 Alternate Test, revision 1 did not adequately verify that MS-41 was properly reinstalled after preventative maintenance. The performance finding is considered more than minor, since the improperly installed valve plug would not have been identified prior to auxiliary feedwater system operability, had it not been identified during an unrelated tagout on the steam supply to the 22 auxiliary feedwater turbine. This is considered very low risk significance in accordance with NRC MC 0609 Appendix G since two alternate core cooling paths were available. This is an example of insufficient Entergy oversight of contractor work activities.
Inspection Report# : 2002007*(pdf)*

Significance: Dec 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE CONFIGURATION CONTROL FOR A SAFETY RELATED SYSTEM
The inspector identified an example of inadequate configuration control for a safety-related system. On November 20, 2002, the inspector identified that two 125 vdc circuit breakers were in their correct position (open) but administrative locking devices were not installed. The breakers are used to cross-connect the 21 and 22 125 vdc buses. This is considered a Non-Cited Violation of Technical Specification 6.8.1.a., which includes requirements for procedure adherence and operations of safety-related systems including the 125 volt DC system. Check off list (COL) 27.1.6, Instrument Buses, DC Distribution and PA Inverter, revision 18 requires the breakers to be open and locked. This performance deficiency is more than minor since more than one breaker was in the required position, but not locked. The finding impacts the mitigating systems cornerstone and is associated with pre-event human error. The finding is considered very low safety significance (Green) since the operability or availability of the 21 and 22 DC buses were not impacted.
Inspection Report# : 2002007*(pdf)*

Significance: Jul 19, 2002
Identified By: NRC
Item Type: VIO Violation
VIOLATION OF THE APPROVED FIRE PROTECTION PROGRAM/THREE-HOUR RATED WALL CONSTRUCTED TO SEPARATE THE CONTROL BUILDING FROM THE TURBINE BUILDING
WHITE - The team identified a violation of License Condition 2.K of Facility Operating License DPR-26. License Condition 2.K requires that Entergy implement and maintain in effect all provisions of the NRC approved fire protection program, which states that a three-hour rated wall will be constructed to separate the control building from the turbine building. In 1978, to meet the three-hour rating, the wall was to have been built in accordance with the design specification Underwriters Laboratories (UL) U902. Contrary to the above, in February 2002, the wall was

found not to be constructed in accordance with UL U902.

The combined effect of the identified deficiencies was that, as of February 2002, passages existed through both the outer brick and inner portions of the wall. If a significant amount of smoke and gasses were to penetrate the wall, this could result in the CCR becoming uninhabitable, causing the operators to resort to using the Alternate Safe Shutdown System. These conditions did not represent a three-hour fire barrier. The NRC risk assessment, using Phase 2 of the NRC Fire SDP described in MC 0609, Appendix F, considered the wall a moderately degraded fire barrier having low to moderate safety significance (White). Until repairs could be completed, Entergy established a compensatory fire watch in accordance with the IP2 fire protection program.

Entergy actions in identifying original construction deficiencies in the CCR west inner wall in February 2002 were commendable. However, the corrective actions taken were not fully effective in restoring the wall to its three-hour rated design configuration. Additionally, the initial extent of condition was not sufficient to identify other degraded fire barrier walls.

[Final Significance Determination and Notice of Violation docketed in NRC letter, dated November 8, 2002. Entergy response to NOV dated December 9, 2002]
Inspection Report# : 2002010*(pdf)*

Barrier Integrity

Significance: G Feb 14, 2003
Identified By: NRC
Item Type: FIN Finding
Ineffective tracking of Inservice Testing of component cooling water system relief valve.
The team identified a finding regarding the scheduled inservice test (IST) of a Component Cooling Water (CCW) system pressure relief valve that was inadvertently not performed during the last plant refueling outage. Access and testing of this valve normally requires the plant to be shutdown.

This finding was not a violation of applicable technical specifications for IST because the timing of the team's questions identified this issue to Entergy personnel while the test was within the allowed test interval extension. However, the issue was more than minor because, if left uncorrected, improperly tracked relief valve tests could result in a more significant safety concern because the valves would not be tested as required to ensure their reliable operation to provide CCW piping over-pressurization protection during accident conditions and maintain the CCW containment penetration barrier integrity. The finding was determined to be of very low safety significance (Green) because there was no actual open pathway in the physical containment structure. (Section 1R21b. 1.1)
Inspection Report# : 2003004*(pdf)*

Significance: G Dec 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
OPERATORS DEVIATE FROM PLANT OPERATING PROCEDURES
On November 23, 2002, during a plant cooldown, Entergy deviated from the guidance of plant operating procedure (POP) 3.3, Plant Cooldown, Rev. 57. The consequence of the failure to follow the POP guidance was to exceed the operational limits on the steam generator tube sheet differential pressure of 1600 psid with a maximum value of

approximately 1855 psid. Control room operators were unaware of this operational limit. Reviews of steam generator manufacturer specifications and the Updated Final Safety Analysis Report design basis accident analysis information indicated that the steam generator tubes were designed to withstand up to 2485 psid during upset and hydrostatic conditions. Therefore, the structural integrity and qualification of the steam generator tubes was maintained. Failure to document the basis of marking non-conditional steps in POP 3.3 as not-applicable is considered a Non-Cited Violation of 10 CFR 50 Appendix B, Criterion V. This issue was considered more than minor because it represented a lack of understanding of procedure requirements and awareness of plant operational limitations. This finding is considered very low safety significance (Green) in accordance with manual chapter 0609 Appendix G, in that the core cooling pathway via the steam generators was not impacted.
Inspection Report# : 2002007*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Significance: G May 15, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
FAILURE TO COMPLY WITH PACKAGING PROCEDURES
A self-revealing non-cited violation of 10 CFR 71.12 was identified for failure to comply with shipping cask package procedures. On February 6, 2003, a CNS 8-120 B cask was received from the Indian Point Energy Center at a consolidation facility in South Carolina with a bolt missing on the primary lid's pressure test port in violation of the cask use and maintenance procedures. This finding was more than minor in that it was associated with the Public Radiation Safety Cornerstone's attribute of procedures for transportation packages. The finding affected the associated cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive materials contained in an NRC-approved Type B package released into the public domain. The finding was determined to be of very low safety significance in that the finding did not involve exceeding transportation radiation limits, there was no breach of the package during transit, and the issue was a Certificate of Compliance maintenance/use performance deficiency.
Inspection Report# : 2003003*(pdf)*

Physical Protection

Miscellaneous

Significance: N/A Feb 03, 2003
Identified By: NRC
Item Type: FIN Finding
Generally effective Corrrective Action Program implementation
The inspectors concluded that, within the scope of the issues reviewed, overall, Indian Point 2 (IP2) personnel were identifying issues at a threshold suitable to recognize conditions adverse to quality and help ensure reliable equipment operation. Although station backlogs (corrective actions, maintenance and engineering items) remained relatively high, the inspectors observed that senior management continued to provide reasonable oversight and emphasis on accountability for corrective action program performance. Corrective action process condition reports adequately characterized and bounded the scope of the problems, and correctly assessed equipment operability. Nevertheless, the team identified instances regarding steam generator level controller replacement problems and a cable tunnel groundwater leak where problems were not identified and not entered into the corrective action process.

IP2 personnel usually evaluated problems to a level of detail appropriate to its technical complexity and risk significance. Problems were adequately prioritized for resolution considering the potential safety significance of the issues and their probability for recurrence. However, in some instances (emergency diesel generator wiring termination and breaker setpoint database), the inspectors identified evaluations where the problems were not completely addressed.

Corrective actions generally addressed the problems and encompassed the scope of the issues. Based on the issues reviewed, the inspectors found corrective actions were scheduled and completed commensurate with the risk significance of the issues. Formal effectiveness reviews were completed, and then reviewed by the Corrective Action Review Board to help ensure the corrective actions were effective in resolving more significant problems. Notwithstanding, corrective actions were not effective to prevent repetitive problems during a steam generator controller replacement modification.
Inspection Report# : 2003002*(pdf)*

Significance: TBD Apr 01, 2000
Identified By: Licensee
Item Type: FIN Finding
Contamination in Storm Drains
Con Edison staff appropriately responded to the discovery of trace amounts of contamination in the Unit 1 storm drains and took proper actions to resolve the condition and to investigate the cause. The material was not associated with the Unit 2 steam generator event or any recent plant activities, and there was no radiological dose consequence due to the contamination.
Inspection Report# : 2000003*(pdf)*
Inspection Report# : 2001010*(pdf)*

Significance: TBD Apr 01, 2000
Identified By: NRC
Item Type: FIN Finding
Steam Generator Tube Leak Root Cause Evaluation
Con Edison completed the investigation of the plant response to the February 15, 2000 steam generator tube leak. Corrective actions to address the causes of weaknesses in the plant response to the event were in progress at the end of the inspection period and NRC review will be the subject of an AIT follow-up team inspection. The results of the root cause investigation for the steam generator tube failure were not reviewed and are being provided by Con Edison to the NRC Office of Nuclear Reactor Regulation for review.
Inspection Report# : 2000003*(pdf)*

Last modified : December 01, 2003