Indian Point 2
1Q/2007 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE RISK ASSESSMENT FOR 21 MBFP STEAM INLET VALVE
The inspectors identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (CFR), Part 50.65(a)(4), because Entergy did not adequately assess and manage the risk of on-line maintenance activities while operating with a degraded steam inlet valve on one of Entergy's two main boiler feed pumps (MBFP). Specifically, from November 16 through 21, 2006, the degraded condition of the 21 MBFP increased the likelihood of a reactor trip, but was not assessed or included in the plant's on-line risk model. Entergy entered this issue into their corrective action program and properly assessed 21 MBFP risk on November 21, 2006.

The inspectors determined that this finding was more than minor because Entergy failed to consider risk significant structures, systems, components, and support systems that were unavailable during the performance of on-line maintenance. Specifically, Entergy failed to assess the increase in online risk from the increased likelihood of a reactor trip due to the 21 MBFP degraded condition. The inspectors evaluated this finding using IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," and determined that this finding was of very low safety significance because the finding resulted in an increase in the incremental core damage probability of less than 1x10-6 (actual increase was approximately 2x10-8).

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not provide complete and accurate procedures, in that, the online risk assessment procedure did not require degraded equipment that impacted risk to be assessed or managed.
Inspection Report# : 2006005 *(pdf)*

Significance: G Sep 30, 2006
Identified By: Self-Revealing
Item Type: FIN Finding
INADEQUATE OPERATING PROCEDURES FOR LOSS OF BOTH HEATER DRAIN TANK PUMPS
A Green self-revealing finding was identified because Entergy failed to develop adequate procedures for governing the response to a loss of both heater drain tank pumps and to an approaching rod insertion limit (RIL) alarm condition. Specifically, the procedure governing operator actions during a loss of heater drain tank pumps did not specify for the operators to reset the steam dumps following the rapid downpower. The alarm response procedure for the approaching rod insertion limit condition directed the operators to place the rod control system in manual to stop further automatic inward rod motion. This impacted operators ability to add negative reactivity and control the transient. Entergy entered these procedural deficiencies into their corrective action program and is evaluating the appropriate steps to correct the procedural deficiencies.

The inspectors determined that this finding is greater than minor because it is associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions. Specifically, the procedural inadequacies complicated operator actions to a rapid downpower, resulted in a manual reactor trip when the operators determined that they did not have sufficient control of the transient, and could impact other accident sequences requiring negative reactivity addition. The inspectors evaluated this finding using Phase I of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined it to be of very low safety significance because it did not contribute to the likelihood of both a reactor trip and the likelihood that mitigation equipment or functions would be unavailable. The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that plant operating procedures were adequate to ensure operators could appropriately

respond to a rapid downpower transient.
Inspection Report# : 2006004 *(pdf)*

Significance: Sep 30, 2006
Identified By: Self-Revealing
Item Type: FIN Finding
INADEQUATE PROCEDURE FOR CALIBRATING THE STEAM DUMP LOSS OF LOAD CONTROLLER
A Green self-revealing finding was identified because Entergy failed to develop an accurate procedure for calibration of the steam dump loss of load controller. This resulted in the steam dumps failing to operate properly during a plant transient, complicating operator response, and leading to a manual reactor trip. Following identification of the issue, Entergy entered the issue into the corrective action program, corrected the procedural deficiency, and re-calibrated the controller.

The inspectors determined that this finding is greater than minor because it is associated with the Procedural Quality attribute of the Initiating events cornerstone; and, it impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions. Specifically, the inadequacy in Entergy's calibration procedure caused the steam dumps to operate improperly during a plant transient and contributed to a reactor trip. The inspectors evaluated this finding using Phase I of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined it to be of very low safety significance because it did not contribute to the likelihood of both a reactor trip and the likelihood that mitigation equipment or functions would be available. The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that the procedure for calibration of the steam dump loss of load controller was accurate, in that, it specified incorrect settings for the controller.
Inspection Report# : 2006004 *(pdf)*

Significance: Jun 30, 2006
Identified By: Self-Revealing
Item Type: FIN Finding
INADEQUATE PROCEDURE FOR PLACING STANDBY MAIN LUBE OIL COOLER IN SERVICE
A Green self-revealing finding was identified because Entergy’s procedure for placing the standby main lube oil cooler in service was inadequate. A deficiency in the procedure resulted in a loss of main feedwater, an automatic start of the motor-driven auxiliary feedwater pumps, and a steam generator level transient. This issue was entered into the corrective action program, and the procedural deficiencies were resolved.

The inspectors determined that this finding was associated with the Initiating Events cornerstone; and, it was more than minor because it was similar to IMC 0612, Appendix E, “Examples of Minor Issues,” Example 4.b, since the inadequacies in Entergy’s procedure caused a plant transient. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” and determined that the finding was of very low safety significance because it did not contribute to the likelihood of both a reactor trip and the likelihood that mitigation equipment or functions would be unavailable. The inspectors also determined that the finding had a cross-cutting aspect in the area of human performance because Entergy’s procedures were not complete and accurate, in that, they failed to ensure the standby main lube cooler was properly filled and vented prior to being placed in service.
Inspection Report# : 2006003 *(pdf)*

Significance: Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE PROCEDURE FOR PLACING RHR PUMP SUCTION PRESSURE GAUGES IN SERVICE
The inspectors identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because Entergy’s procedures failed to ensure that the 22 residual heat removal (RHR) pump suction pressure gauge was placed in service prior to starting the system in the shutdown cooling mode of operation. This gauge, which is used to identify degrading RHR pump performance when in shutdown cooling, was left isolated after the plant was depressurized. Entergy placed the pressure gauge in service and entered the issue into the corrective action program.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. The inspectors evaluated the significance of this finding using IMC 0609, Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs [Pressurized Water Reactors] and BWRs [Boiling Water Reactors] and determined that this finding was of very low safety significance because the finding did not degrade the equipment, instrumentation, training or procedures needed for any shutdown safety function. The inspectors also determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that the procedure for placing the RHR system in the shutdown cooling mode of operation was complete and accurate.
Inspection Report# : 2006003 *(pdf)*

Significance: Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO FOLLOW PLANT PROCEDURES FOR IMPLEMENTATION OF COMPENSATORY MEASURES
The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because plant procedures were not followed during the installation of compensatory measures to restore operability of the RHR pumps following the identification of service water piping degradation in the primary auxiliary building. The inspectors also identified multiple deficiencies with the installation and implementation of the compensatory measures. In response, Entergy corrected the deficiencies associated with the compensatory measures and entered the issue into the corrective action program.

The inspectors determined that this finding, which was associated with the Mitigating Systems cornerstone, was more than minor because it was similar to IMC 0612, Appendix E, “Examples of Minor Issues,” Example 3.a, in that, the deficiencies identified with Entergy’s compensatory measures required significant rework to ensure RHR pump operability. The inspectors evaluated the significance of this finding using IMC 0609, Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs,” Checklist 2, and determined that the finding was of very low significance because the finding did not degrade the equipment, instrumentation, training, or procedures needed for any shutdown safety function. The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not follow plant procedures when implementing a temporary alteration required for the operability of safety-related equipment.
Inspection Report# : 2006003 *(pdf)*

Significance: Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE PROCEDURE FOR VENTING THE REACTOR VESSEL HEAD WHILE SHUTDOWN
The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because plant procedures for reactor coolant system venting following depressurization were inadequate. This resulted in the formation of an 850 gallon void in the reactor vessel head while the plant was shutdown and depressurized. Entergy entered this issue into the corrective action program for evaluation.

The inspectors determined that this finding, which was associated with the Initiating Events cornerstone, was more than minor because if it was left uncorrected, it would have become a more significant safety concern. The inspectors evaluated the significance of this finding using IMC 0609, Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs,” Checklist 3, and determined that a Phase 2 analysis was needed. The Region I Senior Reactor Analyst performed the Phase 2 analysis using IMC 0609, Appendix G, Attachment 2, “Phase 2 Significance Determination Process Template for PWR During Shutdown,” and determined that the finding was of very low safety significance based upon the availability of mitigating systems and the low initiating event (loss of inventory) likelihood. The inspectors also determined that this finding had a cross-cutting aspect in the area of human performance because Entergy’s shutdown procedures were not complete and accurate, in that, they failed to ensure the reactor vessel head was adequately vented.
Inspection Report# : 2006003 *(pdf)*

Mitigating Systems

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE DESIGN CONTROL ASSOCIATED WITH VORTEXING AND NET POSITIVE SUCTION HEAD CALCULATIONS
The team identified a finding of very low significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” in that, Entergy did not ensure adequate suction submergence for the three safety injection (SI) pumps by not properly translating vortex and net positive suction head (NPSH) design parameters into calculations relative to reactor water storage tank (RWST) level. Specifically, Entergy used a non-conservative method to calculate the level required to prevent pump vortexing, and used a non-conservative RWST level value for determining available NPSH for the SI pumps. Entergy entered the issue into their corrective action program and revised the affected calculations.

The finding is more than minor because the calculation deficiencies represented reasonable doubt on the operability of the SI pumps, even though the pumps were ultimately shown to be operable. The finding is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the significance determination process (SDP), documented in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because it was a design deficiency that did not result in a loss of SI system operability, based upon the team’s verification of Entergy’s revised calculations.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE DIFFERENTIAL PRESSURE VALUE USED FOR MOV 746 AND MOV 747 TONENSURE VALVE CAPABILITY
The team identified a finding of very low significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” in that, Entergy did not accurately incorporate design parameters into valve thrust calculations for motor operated valve (MOV) 746 and MOV 747. Specifically, Entergy used an incorrect and non-conservative differential pressure in the calculations for MOV 746 and MOV 747, which were developed to verify that the valves could develop sufficient thrust to open under postulated design basis conditions. Additionally, an incorrect equation was used in determining the reduction in motor torque due to degraded voltage conditions. Entergy entered the issue into their corrective action program and revised the affected calculations using the correct information.

The finding is more than minor because the calculation deficiencies represented reasonable doubt on the operability of MOV 746 and MOV 747. The finding is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it was a design deficiency that did not to result in a loss of MOV 746 and MOV 747 operability, based upon the team’s verification of Entergy’s revised calculations.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE DESIGN CONTROL FOR ENVIRONMENTAL EFFECTS TO ENSURE THE AVAILABILITY OF THE TURBINE DRIVEN AUXILIARY FEEDWATER PUMP OPERATION
The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” in that, Entergy did not establish adequate design control measures to ensure the availability of the turbine driven auxiliary feedwater pump (TDAFWP) during a postulated loss-of-offsite power (LOOP) event. Under certain LOOP situations, the team determined that the TDAFWP steam supply could be inadvertently isolated

because of inadequate calculations and procedures for limiting the AFWP room temperature rise. Specifically, a calculation to determine the auxiliary feedwater pump (AFWP) room temperature rise during a LOOP did not include heat input from the TDAFWP. Further, actions that could limit the rise in AFWP room temperature and prevent the inadvertent isolation of the TDAFW pump (opening an AFWP room roll-up door or promptly restoring forced ventilation) were not included in procedures. Entergy entered this issue into their corrective action program, implemented immediate compensatory actions, and revised AFWP room temperature rise calculations.

The finding is more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of safety function of the TDAFWP (single train) for greater than its 72 hour technical specification allowed outage time, based on the team's review and assessment of site ambient temperature data over the last year.
Inspection Report# : 2007007 *(pdf)*

Significance: Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO ADEQUATELY MONITOR GAS TURBINE SYSTEM PERFORMANCE AS REQUIRED BY THE MAINTENANCE RULE
The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50.65(a)(1), the Maintenance Rule, in that, Entergy failed to monitor the gas turbine (GT) system in a manner that provided reasonable assurance that the system could perform its intended safety function. Specifically, Entergy did not establish appropriate GT reliability goals, and therefore did not take corrective actions, when GT-1 had exceeded these goals for maintenance preventable functions failures (MPFF). In addition, Entergy did not properly classify repeat MPFFs, which resulted in a similar failure to take corrective actions as required. This resulted in additional GT-1 out of service time that would not have happened if appropriate actions had been taken. Entergy entered this issue into their corrective action program and lowered the allowable goal for MPFFs, and revised the GT-1 (a)(1) action plan to improve reliability.

The finding is more than minor because appropriate GT reliability goals were not established commensurate with safety and appropriate corrective actions were not taken when goals were not met. This finding is associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 and Phase 2 of the SDP, which considered that the additional GT-1 out of service time due to this issue could be as much as three days. The finding has a cross-cutting aspect in the area of human performance because Entergy did not adequately ensure procedures were complete, accurate, and up-to-date. Specifically, procedure ENN-DC-171, "Maintenance Rule Monitoring," did not provide steps to discriminate between the classification of an initial design deficiency and further failures due to the same condition, resulting in mis-classifying several GT functional failures.
Inspection Report# : 2007007 *(pdf)*

Significance: Feb 16, 2007
Identified By: NRC
Item Type: FIN Finding
FAILURE TO CORRECT DEGRADED GAS TURBINE 1 RELIABILITY
The team identified a finding of very low safety significance involving Entergy procedure, EN-LI-102, "Corrective Action Process," in that, Entergy failed to take corrective actions to address degraded GT-1 reliability. This resulted in a two and one half day time period in January 2007 when GT-1 and GT-3 were simultaneously inoperable because, after GT-3 was made inoperable for planned maintenance activities, GT-1 was subsequently found to be inoperable. Specifically, the reliability of GT-1 declined from an average of 75% for 2005 and the first 10 months of 2006, to 50% for the three months from November 2006 to January 2007; however, Entergy did not take actions to correct this degraded reliability. Entergy entered this issue into their corrective action program and developed an action plan to address GT reliability issues.

The issue is more than minor because it is associated with the equipment reliability attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on

Phase 1 and Phase 2 of the SDP, assuming that both GT-1 and GT-3 were unavailable for the two and one half days, due to this issue. The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy did not correct degraded reliability of GT-1, resulting in having GT-1 and GT-3 simultaneously inoperable.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE STATION BATTERY CAPACITY TESTING FOR DEGRADATION MONITORING
The team identified a finding of very low safety significance (Green) involving a non-cited violation of Technical Specification 3.8.6.6, in that, Entergy did not perform station battery capacity testing in accordance with IEEE Standard 450-1995 (related to battery maintenance and testing). Specifically, Entergy procedurally terminated battery capacity testing at the rated discharge time (four hours), before reaching the minimum voltage, as specified by IEEE Standard 450-1995. This prevented accurate quantitative measurement of capacity degradation and identification of the need to conduct potential accelerated battery testing, as specified by both IEEE Standard 450-1995 and the technical specifications, if battery capacity drops by more than 10% relative to the previous test. Entergy entered the issue into their corrective action program and performed calculations using past test data, which demonstrated that the capacities of station batteries had not degraded more than 10%.

This issue is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of station battery safety function, based upon the team's verification of Entergy's calculations.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INEFFECTIVE CORRECTIVE ACTION FOR HIGH INTER-TIER BATTERY RESISTANCES
The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," in that, Entergy did not take effective corrective actions for a condition adverse to quality concerning out-of-tolerance inter-tier resistances on the No. 21 station battery. Specifically, after repeated failures of the No. 21 station battery inter-tier resistance testing, vendor and IEEE Standard 450-1995 recommended corrective actions were not taken to correct the adverse out-of-tolerance resistance trend. Entergy entered the issue into their corrective action program and performed calculations, which demonstrated that the voltage drop due to the as-found resistance of the inter-tier connections was small and did not impact No. 21 battery operability.

This issue is more than minor because if it was left uncorrected, it would have become a more significant safety concern. Specifically, high resistance connections in a battery that is loaded during accident conditions can cause localized heating and can cause permanent damage to the battery. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of No. 21 station battery safety function, based upon the team's verification of Entergy's revised calculations. The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy did not take effective corrective actions to address the adverse trend of out-of-tolerance inter-tier resistances.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
UNTIMELY CORRECTIVE ACTIONS FOR DECREASE IN BATTERY MARGIN
The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," in that, Entergy did not promptly identify and correct a condition adverse to quality, with respect to known errors in the No. 23 station battery design calculations. Specifically, Entergy did not recognize at the appropriate time the need to write a condition report, perform an operability determination, or place controls on the use of

the No. 23 battery design calculations when errors were discovered in the No. 23 battery design calculations that significantly lowered the battery capacity margin. Entergy entered the issue into their corrective action program and performed calculations, which demonstrated No. 23 station battery operability through the next refueling outage, based on the calculated margin and conservatisms available.

This issue is more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of No. 23 station battery safety function, based upon the team's verification of Entergy's revised calculations.

The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy failed to promptly identify the decrease in margin found in the No. 23 battery design calculations of record.
Inspection Report# : 2007007 *(pdf)*

Significance: G Dec 31, 2006
Identified By: NRC
Item Type: FIN Finding
FAILURE TO IMPLEMENT CORRECTIVE ACTIONS TO CORRECT A DEGRADED CONDITION WHICH IMPACTED GAS TURBINE #1 RELIABILITY AND AVAILABILITY
The inspectors identified a Green finding, in that, Entergy's corrective actions were inadequate to resolve a deficiency associated with the gas turbine 1 (GT-1) starting diesel. This deficiency was identified following a failure of GT-1 to start on February 7, 2005, and resulted in three subsequent failures. A corrective action was written to correct the deficient condition following the initial failure and was closed on June 22, 2005, with no actions taken based on a senior management decision to cancel preventive maintenance activities on the gas turbines due to pending system retirement. Entergy entered this issue into their corrective action program and installed a modification to the coolant system to prevent further trips due to this condition.

The inspectors determined that this finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, it impacted GT-1 reliability, in that, the deficiency resulted in multiple failures to start on demand after the condition was identified and the action to correct the condition was closed without being implemented. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that a Phase 2 evaluation was required because the finding represented an actual loss of safety function of a non-Technical Specification required train of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours. The inspectors used the Risk-Informed Inspection Notebook for Indian Point Nuclear Generating Unit 2, to conduct the Phase 2 evaluation. The inspectors determined that 65 hours of unavailability were caused by the additional failures of GT-1 due to the starting diesel coolant system deficiency. The inspectors conservatively equated this cumulative unavailability time to the total exposure time and used an initiating events likelihood of less than three days. The Phase 2 approximation yielded a result of very low safety significance (Green).

The inspectors determined this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that equipment and resources were available and adequate to assure reliable operation of GT-1. Specifically, Entergy did not minimize long-standing equipment issues and maintenance deferrals associated with the gas turbine system.
Inspection Report# : 2006005 *(pdf)*

Significance: G Dec 05, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO IDENTIFY A DEGRADED CONDITION OF AN AUXILIARY FEED WATER CHECK VALVE IN THE CORRECTIVE ACTION PROGRAM
The inspectors identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," in that, Entergy failed to identify a condition adverse to quality associated with improper internal clearances on BFD-68, an auxiliary feedwater check valve, in the corrective action program. Specifically, upon inspection in September 2006, the

gasket between the valve's body to bonnet seal was found over-crushed causing the gasket to partially unwind, potentially impacting valve operation. Gasket damage was noted in work orders during internal valve inspections of BFD-68 performed in 1997 and 2002; however, the deficiencies were not identified in the corrective action program. Consequently, the problem was not evaluated and corrected prior to reassembly of the valve. Entergy entered this issue into the corrective action program, evaluated the condition, and conducted repairs to the valve to ensure the proper gasket crush was obtained.

The inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that the finding was of very low safety significance because it was not a design or qualification deficiency; it did not result in the loss of a system safety function or a train safety function for greater than the Technical Specification Allowed Outage Time; and it did not screen as potentially risk significant due to external events.
Inspection Report# : 2006006 *(pdf)*

Significance: G Dec 05, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE EVALUATION OF LEAKING 22 STEAM GENERATOR LOW FLOW BYPASS VALVE FCV-427L
A self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” was identified, in that, Entergy failed to adequately evaluate leakage into the 22 steam generator. During the Indian Point Unit 2 reactor trip on August 23, 2006, main feedwater low flow bypass valve FCV-427L leaked excessively and resulted in an uncontrolled rise in 22 steam generator level; operator response to isolate feedwater to the steam generator in accordance with emergency operating procedures; and automatic actuation of the feedwater isolation system. The excessive leakage condition into the 22 steam generator was identified on April 4, 2006, prior to Indian Point Unit 2 refueling outage 2R17, but was not fully evaluated or corrected prior to the reactor trip on August 23, 2006. This issue was entered into the corrective action program, and FCV-427L was repaired and retested satisfactorily.

The inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of the finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that the finding was of very low safety significance because it was not a design or qualification deficiency; it did not result in the loss of a system safety function or a train safety function for greater than the Technical Specification Allowed Outage Time; and it did not screen as potentially risk significant due to external events.

The inspectors determined that the finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not thoroughly evaluate the cause of excessive leakage into the 22 steam generator such that the resolutions addressed the causes and extent of condition of the problem.
Inspection Report# : 2006006 *(pdf)*

Significance: G Jun 30, 2006
Identified By: Self-Revealing
Item Type: FIN Finding
INADEQUATE CORRECTIVE ACTIONS FOR DEGRADATION OF SERVICE WATER PIPING
A Green self-revealing finding was identified because Entergy failed to take adequate corrective actions for a degraded service water pipe in the primary auxiliary building. Degradation of this pipe was identified in 2003, but was not adequately evaluated or repaired. Consequently, in April of 2006, the continued corrosion of this pipe led to a through-wall leak and, if not corrected, would have challenged the operability of the RHR pumps. Entergy implemented compensatory measures to protect the RHR pumps, repaired the degraded pipe, and entered the issue into the corrective action program.

The inspectors determined that this finding, which was associated with the Mitigating Systems cornerstone, was more than minor because if it was left uncorrected it would have become a more significant safety concern. The inspectors evaluated

the significance of this finding using Phase 1 of IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” and determined that the finding was of very low safety significance because it represented a qualification deficiency that was confirmed not to result in the loss of operability per Part 9900 Technical Guidance, “Operability Determination Process for Operability and Functional Assessment.” The inspectors also determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not implement timely and effective corrective actions for degraded service water piping in the primary auxiliary building.
Inspection Report# : 2006003 *(pdf)*

Significance: G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO IDENTIFY DEGRADED RESIDUAL HEAT REMOVAL PUMP CELL FIRE DOOR
The inspectors identified a Green NCV of license condition 2.K. because Entergy failed to identify a condition adverse to fire protection related to a degraded fire door between the 21 and 22 RHR pump cells. A similar condition with the same door had been previously identified by the NRC in January 2006. Entergy took actions to correct the degraded fire door and entered the issue into the corrective action program.

The inspectors determined that this finding was more than minor because it was associated with the Protection Against External Factors attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using IMC 0609 Appendix F, “Fire Protection Significance Determination Process,” and determined that the finding was of very low safety significance because the fire door, which was moderately degraded, provided a minimum of 20 minutes of fire endurance protection; and, the ignition sources and combustible materials in the RHR pump cells were situated in a manner that the degraded fire door would not have been subject to direct flame impingement. The inspectors also determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because operators who routinely traverse through the degraded fire door during performance of their rounds had not identified the degraded condition of the door.
Inspection Report# : 2006003 *(pdf)*

Significance: G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE POST-WORK TEST ON 21 EMERGENCY DIESEL GENERATOR
The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” because Entergy’s post-maintenance test on the 21 emergency diesel generator (EDG) following a governor replacement in November 2004 was not adequate to ensure it could perform its intended design function. Subsequent testing showed the EDG could not attain its rated load of 2300 kilowatts. Entergy corrected the deficiency with the 21 EDG, performed a post-maintenance test including a run at 2300 kilowatts, and entered the issue into the corrective action program.

The inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, “Significant Determination of Reactor Inspection Findings for At-Power Situations,” and determined that this finding was of very low safety significance because it was not a qualification deficiency; it did not represent a loss of safety function for a train or system as defined in the plant specific risk-informed inspection notebook; and it was not risk significant due to external event initiators.
Inspection Report# : 2006003 *(pdf)*

Significance: G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO ASSESS THE RISK OF MAINTENANCE ACTIVITIES ON VALVE SI-869A
The inspectors identified a Green NCV of 10 CFR Part 50.65(a)(4) because Entergy did not assess the risk associated with maintenance on the discharge containment isolation valve from the 21 containment spray pump, SI-869A. This

maintenance resulted in the unavailability of the 21 containment spray train for a period of approximately 90 minutes. Entergy entered this issue into the corrective action program, conducted an extent of condition review, and completed a causal analysis.

The inspectors determined that this finding, which was associated with the Mitigating Systems cornerstone, was more than minor because it was similar to Example 7.e in IMC 0612, Appendix E, “Examples of Minor Issues,” in that, the licensee’s risk assessment failed to consider maintenance activities on components that prevent containment failure. The inspectors evaluated the significance of this finding using IMC 0609, Appendix K, “Maintenance Risk Assessment and Risk Management Significance Determination Process,” Flowchart 1, and determined that the finding was of very low safety significance because the calculated risk deficit was not greater than 1 x 10-6. The inspectors also determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not appropriately incorporate risk insights into planning work activities on SI-869A in accordance with 10 CFR Part 50.65(a)(4) and the Site Management Manual IP-SMM-WM-101, “Online Risk Assessment.”
Inspection Report# : 2006003 *(pdf)*

Significance: G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE SURVEILLANCE TEST PROCEDURE FOR EMERGENCY DIESEL GENERATORS
The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because plant surveillance procedure 2-PT-R84B, “22 EDG 8 Hour Load Run,” was not adequate to ensure testing at the appropriate power factor limit prescribed by Technical Specifications Surveillance Requirement 3.8.1.10. Entergy entered this issue into the corrective action program and completed an evaluation to assess the operability of all three EDGs.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, “ Significance Determination of Reactor Inspection Findings for At-Power Situations,” and determined that this finding was of very low safety significance because it was not a qualification deficiency; it did not result in the loss of a system or train safety function; and it did not screen as potentially risk-significant due to external events. The inspectors also determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that procedure 2-PT-R84B, “22 EDG 8 Hour Load Run,” was complete and accurate.
Inspection Report# : 2006003 *(pdf)*

Barrier Integrity

Significance: SL-IV Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE CONTAINMENT CLOSURE EQUIPMENT
The inspectors identified a Severity Level IV NCV of 10 CFR 50.59, “Changes, Tests and Experiments,” for failure to obtain a license amendment pursuant to 10 CFR 50.90 prior implementing a change to alter the requirements of a shutdown fission product barrier. The inspectors reviewed Safety Evaluation 04-0732-MD-00-RE R1, “Installation of a Temporary Roll-up Door on the Containment Equipment Hatch,” to determine if the conclusion that a licensee amendment was not required was correct. Entergy concluded that the roll-up door was equivalent to the closure plate and, therefore, adequate to close containment as required by the action statement. The inspectors found that the door was not designed to be air-tight; therefore, any radioactive release inside containment would bypass the roll-up door. The inspectors concluded that the roll-up door did not meet the design or licensing basis of the closure plate as described in the Updated Final Safety Analysis Report (UFSAR) and previously approved license amendments. Consequently, Entergy incorrectly concluded that a license amendment pursuant to 50.90 was not required prior to implementing the change. Entergy entered the issue into their corrective action program to evaluate and correct.

The inspectors determined that Entergy changed the requirements for the shutdown fission product barrier (containment) prior to receiving NRC approval. As a result, traditional enforcement was used to evaluate the issue because the deficiency affected the NRC's ability to perform its regulatory function. The severity level of the violation was determined to be Severity Level IV in accordance with example D.5 of Supplement 1 of the NRC Enforcement Policy. Additionally, the issue was determined to be of very low safety significance (Green) based on the low decay heat levels at the time the roll-up door was credited in accordance with the significance determination process described in Inspection Manual Chapter (IMC) 0609 Appendix H, “Containment Integrity.”
Inspection Report# : 2006005 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Significance: Dec 31, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
FAILURE TO SURVEY AND PROVIDE ACCESS TO AN UNPOSTED HIGH RADIATION AREA
A Green, self-revealing NCV of 10 CFR 20.1501 with respect to 10 CFR 20.1902(b) was identified, in that, Entergy failed to survey radiological condition changes after a plant manipulation that was likely to cause a change in radiological conditions, and this led to the failure to post a plant area as a high radiation area. As a result, two workers were allowed access to an unsurveyed and unposted high radiation area.

The finding is more than minor because it is associated with the Occupational Radiation Safety cornerstone attribute of exposure control and affected the cornerstone objective, because not establishing radiological conditions and commensurate controls after changing plant radiological conditions prior to allowing access to the affected areas can cause increased personnel exposure. The inspectors evaluated this finding using IMC 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process,” and determined that it was of very low safety significance (Green) because it did not involve ALARA planning and controls, an overexposure, a substantial potential for overexposure, or an impaired ability to assess dose. This issue was entered into Entergy's corrective action program and training was provided to the radiation protection staff.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not use a conservative assumption in the decision-making process, in that, the watch radiation protection technician did not question the radiological conditions of the pipe chase area after a change of plant conditions had occurred and did not require a survey of the pipe chase area before authorizing access to personnel.
Inspection Report# : 2006005 *(pdf)*

Significance: Dec 31, 2006
Identified By: Self-Revealing
Item Type: FIN Finding
UNIT 2 CONTAINMENT SUMP STRAINER MODIFICATION COLLECTIVE EXPOSURE OVERRUNS DUE TO INADEQUATE MOD PREPARATION
A self-revealing finding was identified that involved inadequate modification planning and construction preparations relative to a Unit 2 containment sump strainer modification that resulted in significant unplanned collective exposure (93.7 person-rem compared to a work activity estimate of 10.9 person-rem). Specifically, the actual job site conditions for installation of the containment sump modification were not adequately evaluated with respect to the radiological impact of increased occupancy in high dose rate work areas. This unplanned additional in-field high radiation work resulted in significant unintended exposure that could have been avoided. This issue was entered into Entergy's corrective action program so that lessons learned could be incorporated into the Unit 3 containment sump modification.

The inspectors determined that this finding was more than minor because it was similar to examples 6.a and 6.b of IMC

0612, Appendix E, "Examples of Minor Issues," in that, the issue involved actual collective exposure greater than 5 person-rem and was greater than 50 percent above the estimated or intended exposure; and the majority of the dose overrun was due to activities within Entergy's control. The inspectors evaluated this finding using IMC 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process,” and determined that the finding was of very low safety significance (Green) because it involved an ALARA planning issue, and the 3-year rolling average collective dose for Unit 2 was less than 135 person-rem (73 person-rem average annual exposure for 2003 through 2005).

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not adequately incorporate job site conditions in the work control planning process.
Inspection Report# : 2006005 *(pdf)*

Significance: Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE SURVEY DURING CORE BARREL REPLACEMENT CAUSED UNINTENDED EXPOSURE
A Green self-revealing NCV of 10 CFR Part 20.1501, “General,” was identified because Entergy failed to take adequate radiation surveys during the installation of the core support barrel. As a result, Entergy did not recognize that actual radiological conditions were significantly different than expected, which contributed to unplanned and unintended exposure of a worker. Entergy entered this issue into the corrective action program and completed a root cause analysis.

The inspectors determined that this finding was more than minor because it was associated with the Program and Process attribute of the Occupational Radiation Safety cornerstone; and, it affected the cornerstone objective of ensuring adequate protection of workers from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The inspector evaluated the significance of this finding using IMC 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process,” and determined that this finding was of very low safety significance because it did not involve: (1) as low as reasonable achievable planning or work controls; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose.
Inspection Report# : 2006003 *(pdf)*

Significance: Jun 30, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
FAILURE TO IMPLEMENT PROCEDURAL REQUIREMENTS ASSOCIATED WITH CORE SUPPORT BARREL REPLACEMENT
A Green self-revealing NCV of Technical Specification 5.4.1 was identified because Entergy failed to follow procedural requirements during the core support barrel installation activity. As a result, dose rates were significantly higher than expected during the work activity, and a worker received an unplanned and unintended radiation exposure. Entergy entered this issue into the corrective action program and completed a root cause analysis.

The inspectors determined that this finding was more than minor because it was associated with the Program and Process attribute of the Occupational Radiation Safety cornerstone; and, it affected the cornerstone objective of ensuring adequate protection of workers from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The inspectors evaluated the significance of this finding using IMC 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process,” and determined that the finding was of very low safety significance because it did not involve: (1) as low as reasonable achievable planning or work controls; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose. The inspectors also determined that the finding had a cross-cutting aspect in the area of human performance because Entergy personnel failed to comply with plant procedures that were required and specified to support reinstallation of the core support barrel.
Inspection Report# : 2006003 *(pdf)*

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Significance: G Dec 05, 2006
Identified By: NRC
Item Type: FIN Finding
FAILURE TO ENTER SAFETY CULTURE ASSESSMENT RESULTS INTO CORRECTIVE ACTION PROGRAM
The NRC inspectors identified a finding when Entergy failed to initiate condition reports in accordance with EN-LI-102, “Corrective Action Process,” for the adverse conditions identified in the 2006 Safety Culture Assessment. Consequently, the adverse conditions were not evaluated and appropriate corrective actions were not identified in a timely manner. The contractor who performed the independent safety culture assessment presented the site specific results to Entergy management in June 2006. The negative responses and declining trends identified in the assessment constituted adverse conditions that should have been entered into the corrective action program. At the time of the inspection, Entergy had not initiated condition reports for the assessment results. Consequently, the results had not been fully evaluated to understand the causes and identify appropriate actions to address the identified issues. Additionally, organizations identified by the contractor as needing management attention had not developed departmental action plans at the time of the inspection. Entergy entered this issue into the corrective action program and initiated a learning organization condition report to track development and implementation of action plans to address the assessment results.

The inspectors determined that the finding was more than minor because if left uncorrected it would become a more significant safety concern. Without appropriate action, the weaknesses in the safety culture onsite would continue, increasing the potential that safety issues would not receive the attention warranted by their significance. The finding was not suitable for SDP evaluation, but has been reviewed by NRC management and has been determined to be a finding of very low safety significance. The finding was not greater than very low safety significance because the inspectors did not identify any issues that were not raised which had an actual impact on plant safety or were of more than minor safety significance.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not identify issues with the potential to impact nuclear safety in the corrective action process for evaluation and resolution in a timely manner.
Inspection Report# : 2006006 *(pdf)*

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