Indian Point 2
4Q/2006 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 - 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. The degraded steam inlet valve allowed the 21 MBFP to operate at full power only if 22 MBFP was also running. On a loss of 22 MBFP, 21 MBFP would not be able to continue feeding the steam generators as designed due to a not fully open high pressure inlet steam valve, which would result in a reactor trip. Due to the degraded condition of 21 MBFP, Entergy implemented a temporary procedure change that directed operators to trip the reactor if the 22 MBFP stopped while the plant was operating at power. Entergy entered this issue into their corrective action program and properly assessed 21 MBFP risk on December 21, 2006.

The inspectors determined that this finding was more than minor because Entergy failed to consider risk significant SSCs 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 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 2x10-8).

The inspectors determined that this finding had a Human Performance cross-cutting aspect in that procedural inadequacies existed in the online risk assessment procedure because it did not require degraded equipment which impacted online risk to be evaluated. (Section 1R13)

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: G 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: G 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: G 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: G 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: G 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)*

Significance: G Mar 01, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
SCAFFOLDING CONTROL ISSUE RESULTS IN REACTOR TRIP
The NRC identified a Green self-revealing NCV of 10 CFR 50.65(a)(4) because Entergy did not adequately assess the risk associated with scaffold construction activities in the cable spreading room. Entergy procedure IP-SMM-WM-100, "Work Management Process," requires a risk assessment for activities that increase the risk of a plant transient. No risk assessment was completed for this work as part of the work planning process, and as a result, no risk management actions were developed. During scaffold construction, a contractor inadvertently bumped a switch which resulted in 12 dropped control rods and a subsequent manual reactor trip. Entergy entered this issue into the corrective action program and took immediate actions to improve control of scaffold construction activities.

This finding is greater than minor because it was similar to Example 4.b. of IMC 0612, Appendix E, "Examples of Minor Issues," in that the performance deficiency contributed to an actual reactor trip. This finding is of very low safety significance because while it resulted in a reactor trip, it did not also contribute to the unavailability of mitigating systems. The inspectors determined that this finding had a human performance cross-cutting aspect in that Entergy personnel failed to appropriately incorporate risk insights into planning of work activities in close proximity to trip risk components.
Inspection Report# : 2006002 *(pdf)*

Mitigating Systems

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 corrective actions were inadequate to repair 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 senior management decision to preempt preventive maintenance activities on the gas turbines due to pending system retirement. Entergy entered this issue into the 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 since 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 conducted a Phase 1 SDP screening and determined that a Phase 2 evaluation was required since the finding represented an actual loss of safety function a non-Technical Specification required train of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours. 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 in that Entergy did not ensure that equipment and resources were available and adequate to assure nuclear safety. Specifically, Entergy did not maintain plant safety through the minimization of long-standing equipment issues and the minimization of maintenance deferrals associated with the gas turbine system. (Section 1R12)

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)*

Significance: G Mar 01, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO EFFECTIVELY CONTROL THE PERFORMANCE OF THE ROD POSITION INDICATION SYSTEM

The NRC identified a Green NCV of 10 CFR 50.65(a)(2) because Entergy failed to effectively control the performance of the rod position indication system through the use of appropriate preventative maintenance. This resulted in the failure of seven rod bottom lights to illuminate following a reactor trip, creating an additional challenge to plant operators. Entergy entered this issue into their corrective action program and is taking actions to upgrade their surveillance and maintenance procedures relative to the rod position indication system.

The inspectors determined that this finding was greater than minor because it affected the Mitigating Systems cornerstone attribute of Equipment Performance, 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 was determined to be of very low safety significance because it did not result in loss of a system or train safety function and did not screen as potentially risk-significant due to seismic, flooding, or severe weather initiating event. The inspectors determined that the finding had a problem identification and resolution cross-cutting aspect because Entergy did not thoroughly evaluate multiple rod position indication bistable failures such that the resolution addressed the causes and extent of condition of problems.
Inspection Report# : 2006002 *(pdf)*

Significance: G Feb 22, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE CORRECTIVE ACTIONS FOR UTILITY TUNNEL DEGRADATION
The NRC identified a Green self-revealing NCV of license condition 2.K. because Entergy did not take adequate corrective actions for degraded fire protection piping in the utility tunnel. This issue contributed to failure of a 10 inch high-pressure fire protection line in the tunnel. Isolation of this leak resulted in loss of high-pressure fire water to three hose stations in the utility tunnel and three fire hydrants on site. Entergy entered this issue into their corrective action program and is evaluating plans to assess and upgrade the utility tunnel.

This finding is greater than minor because if left uncorrected it would become a more significant safety concern. This finding is of very low safety significance because the areas that lost high-pressure fire water did not contain safety-related or post-fire safe shutdown equipment. The inspectors determined that this finding had a problem identification and resolution cross-cutting aspect because Entergy did not implement timely and effective corrective actions for safety issues associated with degraded piping in the utility tunnel.
Inspection Report# : 2006002 *(pdf)*

Significance: G Jan 29, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
DEGRADED RESIDUAL HEAT REMOVAL PUMP FIRE DOOR
The NRC identified a Green NCV of license condition 2.K. because Entergy failed to identify a degraded three-hour rated fire door between the 21 and 22 residual heat removal pump cells. The door, which provides a barrier to fire and hot gases between the two cells, was determined to be inoperable due to a 3/8 inch gap between the door and frame along the lower half of the door. Entergy entered this issue into the corrective action program and realigned the door.

This finding is greater than minor because it was associated with the Mitigating Systems cornerstone attribute of Protection Against External Factors, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that this finding is of very low safety significance because the degradation of the fire barrier was low, based on the gap in the door having minimal impact on its performance and reliability. The inspectors determined that the finding had a problem identification and resolution cross-cutting aspect because operators who routinely traverse through the degraded fire door during performance of their rounds had not identified the condition of the door in the corrective action system.
Inspection Report# : 2006002 *(pdf)*

Barrier Integrity

Significance: SL-IV Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE CONTAINMENT CLOSURE EQUIPMENT
The team identified a Severity Level (SL) 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 team reviewed Safety Evaluation (SE) 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. The SE was performed to assess the adequacy of using a roll-up door to meet the requirements of the Technical Specification action statements 3.9.4.A.4 and 3.9.5.B.3. The action statements required that the equipment door or closure plate be properly installed within four hours after a loss of decay heat removal. Specifically, 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 team found that the door was not designed to be air-tight, therefore, any radioactive release inside containment would bypass the roll-up door. The team 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 and previous approved license amendments. Entergy entered the issue into their corrective action program to evaluate and correct.

The team found 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 ability to perform its regulatory function. The severity level of the violation was determined to be SL 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 IMC 0609 Appendix H "Containment Integrity" significant determination process. (Section 1R02)

Inspection Report# : 2006005 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Dec 31, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
FAILURE TO SURVEY AND PROVIDE ACCESS TO AN UNPOSTED HIGH RADIATION AREA
On August 27, 2006, a self-revealing NCV of 10 CFR 20.1501 with respect to 10 CFR 20.1902(b) was discovered because the licensee failed to conduct a survey to establish the radiological conditions and commensurate postings and controls of an unposted high radiation area that was affected by known changing plant conditions, prior to allowing personnel access to this area. This finding was entered into the licensee's corrective action program and training was provided.

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 inspector determined that the finding was of very low safety significance (Green) because it did not involve an overexposure, a substantial potential for overexposure, or an impaired ability to assess dose. This finding has a cross-cutting aspect in the area of human performance because the licensee did not use a conservative assumption in the decision-making process, in that the Watch RP 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. (Section 2OS1)

Inspection Report# : 2006005 *(pdf)*

Significance: G 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 discovered 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). The dose overrun was primarily due to inadequate work activity planning. 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 resulted in a significant amount of as-found interferences that required removal and reinstallation and differences in as-found dimensions required a significant amount of fit-up problems which required additional in-field high radiation area work. This unplanned additional in-field high radiation work resulted in significant unintended exposure that could have been avoided. This finding was entered into the licensee's corrective action program including lessons learned for the Unit 3 containment sump modification.

This finding is more than minor because it resulted in unplanned, unintended collective dose that was greater than 50% above the intended dose and greater than 5 person-rem due to conditions that were reasonably within the licensee's ability to foresee and correct and which should have been prevented. The inspector determined that the finding was of very low safety significance (Green) because: the finding was due to ALARA work control planning and the 3-year rolling average collective dose for Unit 2 was less than 135 person-rem (73 person-rem for 2003-2005). This finding has a cross-cutting aspect in the area of human performance because the licensee did not adequately incorporate job site conditions in the work control planning process. (Section 2OS2)

Inspection Report# : 2006005 *(pdf)*

Significance: G 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: G 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)*

Last modified : March 01, 2007