FitzPatrick
1Q/2004 Plant Inspection Findings

Initiating Events

Significance: Sep 25, 2003
Identified By: Self Disclosing
Item Type: FIN Finding
Inadequate corrective action resulted in RWR pump trip and unplanned power reduction.
The inspectors identified a self-revealing finding involving inadequate corrective action for a 1999 reactor water recirculation (RWR) pump trip that resulted in another RWR pump trip and unplanned power reduction on September 25, 2003.

The finding is considered more than minor because it is associated with the equipment performance attribute and resulted in an unplanned plant transient that affected the reactor safety initiating events cornerstone objective of limiting the likelihood of events that upset plant stability. The finding is of very low safety significance because it did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiator, did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, and did not increase the likelihood of a fire or internal/external flood.

Inspection Report# : 2003010*(pdf)*

Mitigating Systems

Significance: Oct 02, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Calculation Assumption for Station Blackout Battery Load Shed not Translated into the Procedure
The team identified a non-cited violation (NCV) regarding the licensee's failure to incorporate the assumptions of the battery loading calculations into the station's operating procedures for a station blackout, as required by 10CFR50, Appendix B, Criterion III, Design Control.

This finding is more than minor since it is associated with the design control 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 (i.e., core damage). The issue was not a design or qualification deficiency that the licensee had evaluated in accordance with GL 91-18, and was determined to be of very low safety significance (Green) because it did not result in an actual loss of safety function of a single train for internal or external event initiated core damage sequences.

Inspection Report# : 2003009*(pdf)*

Significance: Oct 02, 2003
Identified By: NRC
Item Type: FIN Finding
Preconditioning of HPCI Valves Prior to Stroke Time Testing
The team identified that the High Pressure Coolant Injection (HPCI) surveillance procedures failed to test four valves in the as-found condition because the valves were operated at least one time prior to performing the ASME in-service timing test.

This finding is more than minor since it is associated with the procedure quality 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 (i.e., core damage). The issue was not a design or qualification deficiency that the licensee had evaluated in accordance with GL 91-18, and was determined to be of very low safety significance (Green) because it did not result in an actual loss of safety function of a single train for internal or external event initiated core damage sequences.

Inspection Report# : 2003009*(pdf)*

Significance: G Oct 02, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Untimely Replacement of Switches for EDG Output Breaker Cubicles
The team identified a NCV of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, involving the licensee's failure to replace the 52STA switches in three of the four emergency diesel generator (EDG) output breaker cubicles in a timely manner.

This finding is more than minor since it is 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 (i.e., core damage). The issue was not a design or qualification deficiency that the licensee had evaluated in accordance with GL 91-18, and was determined to be of very low safety significance (Green) because it did not result in an actual loss of safety function of a single train for internal or external event initiated core damage sequences.

Inspection Report# : 2003009*(pdf)*

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate corrective actions associated with the failure of RCIC pump discharge flow controller 13FIC-91.
During a reactor core isolation cooling (RCIC) surveillance test on June 10 the system's flow controller did not immediately return system flow rate to its setpoint value after flow decreased when operators manually raised system pressure. The degraded condition was caused by inadequate preventive maintenance that resulted from ineffective corrective action for a similar flow controller problem that occurred in July 2000.

The finding is more than minor, because it affected the mitigating systems cornerstone attribute of equipment performance. The degraded condition of RCIC could have prevented the system from providing adequate flow to the reactor. Therefore, this deficiency affected the reliability and capability of a system that responds to initiating events to prevent undesirable consequences. In accordance with MC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors determined that this finding is of very low safety significance, because it was not a design or qualification deficiency, and it did not result in an actual loss of safety function for the RCIC system with respect to internal or external events.

Inspection Report# : 2003008*(pdf)*

Significance: G Jul 31, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate and untimely corrective action for a 1996 RHR pump discharge check valve failure resulted in a similar failure in Oct 2002 and a violation of 10CFR50 Appendix B Criterion XVI.
The inspector identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI concerning the failure of the D residual heat removal (RHR) pump discharge check valve during the October 2002 refueling outage. This failure resulted due to inadequate corrective action for a similar December 1996 A RHR pump discharge check valve disk hangar arm failure and involved an inappropriate deferral of actions and planned engineering work which was lost track of.

This finding is more than minor because it impacted the mitigating systems cornerstone objective of ensuring the availability and reliability of mitigating systems. During the refueling outage the failure of the D RHR pump discharge check valve could have prevented the B RHR train from performing its shutdown cooling safety function. At the time of the finding the plant was in the refuel mode, with both RHR shutdown cooling systems out of service for maintenance and the decay heat removal system in service. In accordance with NRC Manual 609, Appendix G, "Shutdown Operation Significance Determination Process," the finding is considered to be of very low safety significance because the shutdown cooling safety function was not significantly degraded.
Inspection Report# : 2003005*(pdf)*

Barrier Integrity

Significance: G Dec 31, 2003
Identified By: NRC
Item Type: NCV NonCited Violation

Inadequate procedure for isolation of control room ventilation during a LOCA.
The inspectors identified a non-cited violation of 10CFR 50, Appendix B, Criterion III, "Design Control," that requires regulatory requirements and the design basis to be correctly translated into procedures. Entergy revised an abnormal operating procedure such that isolation of the control room envelope following a loss of coolant accident (LOCA) would not be initiated as analyzed in the design basis control room habitability calculation described in the UFSAR.

The finding is more than minor because it is associated with the procedure quality and adequacy attribute and affected the objective of the reactor safety barrier integrity cornerstone to provide reasonable assurance that physical design barriers protect control room operators from radiological releases caused by accidents. The finding was of very low safety significance because it represented only a degradation of the radiological barrier function provided for the control room, and the increased operator dose would not have exceeded regulatory limits.

Inspection Report# : 2003010*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Miscellaneous

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