Davis-Besse
1Q/2005 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
UNANTICIPATED PLANT RESPONSE TO LARGE BORIC ACID ADDITION TO RCS
A finding of very low safety significance was self-revealed when the control room operators did not demonstrate conservative actions when they failed to fully anticipate the plant response to a large boric acid addition to the reactor coolant system, which was conducted as part of the planned de-boration of number 2 mixed bed demineralizer. The resulting transient caused the controlling control rod group to move to its full out position and required operator to take manual action to decrease the Unit Load Demand until all of the demineralized water was added, which allowed the control rod index to return to normal position, and power was reduced approximately 1.5 percent as a result of this action. The primary cause of this finding was related to the cross-cutting area of Human Performance because the control room operators failed to operate the plant in a controlled manner, as required by plant procedures. The inspectors determined that the finding was more than minor because it directly involved the human performance attribute of the Initiating Events cornerstone's objective which is to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding was of very low safety significance because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available if called upon. This issue was a Non-Cited Violation of Technical Specification 6.8.1, which required, in part, to establish and implement procedures that provide guidance on authorities and responsibilities for safe operation and shutdown of the reactor plant.
Inspection Report# : 2005005*(pdf)*

Significance: G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
REACTOR OPERATOR ATTEMPTED TO ADD WATER TO THE MAKE UP TANK UTILIZING EQUIPMENT THAT WAS OUT OF SERVICE
A finding of very low safety significance was self-revealed when control room staff attempted to add water to the makeup tank using equipment that had been removed from service as part of the clearance which supported work on makeup system valves MU362 and MU363. The control room staff was unaware of the status of the normal makeup water sources to the reactor coolant system, even though the system's status was clearly documented in the Limiting Condition for Operation Tracking Log, a document which is required to be reviewed by the Shift Manager, the Unit Supervisor, and the Reactor Operator prior to shift turnover. The inspectors concluded that the finding was more than minor because the operator's lack of knowledge of system status challenged their ability to adjust control rod index by adding water to the reactor coolant system and to perform selected abnormal operating procedures prepared to address small reactor coolant system leaks. This finding was of very low safety significance because, during the time period the clearance impacted the operation of the makeup water sources, neither the ability to control makeup tank water level or to maintain an appropriate rod control index were challenged. This was determined to be a Non-Cited Violation of Technical Specification 6.8.1.a.
Inspection Report# : 2004014*(pdf)*

Mitigating Systems

Significance: G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
LICENSEE'S DECAY HEAT SYSTEM RESTORATION FROM PLANNED MAINT PERMITTED FORMATION OF AIR VOID IN DECAY HEAT LINE FROM RCS TO DECAY HEAT PUMP #2
A finding of very low safety significance was self-revealed when licensee personnel, during review of the configurations of decay heat piping, determined that recent system restoration from the decay heat pump 2 seal refurbishment was not adequate to prevent the formation of an air void in the decay heat line from the reactor coolant system to the decay heat pump 2. An air void in the line could impede operator's efforts to establish decay heat cooling and, if required, post loss of coolant accident boron precipitation control using the decay heat system. The primary cause of the finding was related to the cross cutting area of Human Performance because the preparers and reviewers of the system clearance for the pump seal refurbishment failed to identify that the vent path specified in the system restoration was not the high point of the piping that had been drained and that another vent path could be made available that would have precluded an air void formation in the piping during

system refill. The inspectors determined that the finding was more than minor because it directly involved the human performance attribute of the Mitigating System cornerstone's objective which is to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance because the finding did not result in any actual loss of safety function and did not screen as significant using the criteria as outlined in the mitigating system section of the Phase 1 significant determination worksheet. This issue was a Non-Cited Violation of Technical Specification 6.8.1, which required, in part, the development and implementation of procedures that provide guidance on equipment control and instructions for filling and venting the decay heat cooling system.
Inspection Report# : 2005005*(pdf)*

Significance: G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
UNTIMELY CORRECTIVE ACTIONS TO CORRECT SAFETY-RELATED BATTERY TERMINAL CONNECTION CORROSION
The inspectors identified a finding of very low safety significance for the licensee's failure to take timely corrective action to remove visible corrosion on several terminal connections on the station's safety-related 2P and 2N batteries. The primary cause of this finding was related to the cross-cutting area of problem identification and resolution. The inspectors determined that the finding was more than minor because it impacted the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This issue was determined to be of very low safety significance because there were no documented cases of any of the affected battery cell's terminal resistance measurements in excess of 150 micro-ohms. This was considered to be a Non-Cited Violation of 10CFR 50, Appendix B, Criterion XVI.
Inspection Report# : 2004016*(pdf)*

Significance: G Dec 16, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO ADEQUATLEY AND EFFECTIVELY CORRECT DEGRADATION OF UNDERGROUND CABLES DUE TO WATER INTRUSION
The inspectors identified a finding of very low significance associated with a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to assure that adequate and timely actions were promptly identified and implemented to address several underground wetted cable issues, a condition adverse to quality. This inspector-identified issue was greater than minor because if left uncorrected, the issue could become a more significant safety concern and could affect the mitigating systems attributes of equipment performance reliability. The inspectors evaluated the finding using Inspection Manual Chapter 0609, "Significance Determination Process," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening. The inspectors determined that the finding (1) did not result in a design or qualification deficiency confirmed to result in a loss of function per NRC Generic Letter 91-18; (2) did not represent an actual loss of safety function; (3) did not result in a loss of safety function of a single train for greater than the Technical Specification allowed outage time; (4) did not represent an actual loss of safety function of one or more non-Technical Specification trains designated as risk significant per the Maintenance Rule for greater than 24 hours; and (5) did not screen as potentially risk significant due to seismic, fire, flooding, or severe weather initiating events.
Inspection Report# : 2004017*(pdf)*

Significance: G Dec 16, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO INITIATE A CONDITION REPORT FOR CONDITIONS ADVERSE TO QUALITY IN THE AUXILIARY FEEDWATER SYSTEM
The inspectors identified a finding of very low significance associated with a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings, for failure to initiate a condition report for conditions adverse to quality in the auxiliary feedwater system. The inspectors identified two conditions adverse to quality that had not been entered into the corrective action program. The inspectors determined that the failure to initiate condition reports for conditions adverse to quality in the auxiliary feedwater system was greater than minor because if left uncorrected the issue would become a more significant safety concern involving programmatic and equipment issues. The inspectors determined that the finding was not suitable for SDP evaluation because the failure to initiate the condition reports did not directly result in degraded or inoperable equipment. Therefore, this finding was reviewed by Regional Management, in accordance with IMC 0612 Section 05.04c, and determined to be of very low safety significance.
Inspection Report# : 2004017*(pdf)*

Significance: G Nov 13, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO COMPLY WITH TS REQUIRED PROCEDURES FOR CONTROL OF STATUS OF CCW 1 BREAKER WHILE DANGER TAGGED

A finding of very low safety significance was self-revealed when maintenance personnel failed to adhere to the requirements of administrative work control documents and instructions during maintenance activities associated with the racking in of breaker AC113 [CCW 1 Motor Supply Breaker]. With a danger tag hanging on the breaker cubicle door, licensee personnel installed the breaker into the breaker cubicle with the intention of operating the breaker to perform maintenance checks on the CCW pump 1 motor. This was contrary to licensee's procedural requirements. The inspectors concluded that the finding was more than minor because, if left uncorrected, it could become a more significant safety concern. Specifically, the operation or manipulation of the danger tagged AC-113 breaker could have resulted in equipment damage or serious personnel injury. This issue was determined to be of very low safety significance because there was no maintenance evolutions in progress on the CCW 1 pump motor which would have required the breaker AC-113 to be racked out at the time of the breaker was inappropriately racked into its cubicle. This was determined to be a Non-Cited Violation of Technical Specification 6.8.1.a.
Inspection Report# : 2004015*(pdf)*

Significance: SL-IV Nov 13, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO PERFORM APPROPRIATE 50.59 SCREENING CONCERNING ECCS SUMP PUMP AFFECT ON ECCS COMPONENT OPERABILITY
A failure to perform an adequate safety evaluation review as required by 10 CFR 50.59 for changes made to the facility as described in the Updated Safety Analysis Report (USAR) was self-revealed. In June 2003, the licensee changed the limit and precautions sections of several procedures to specify that emergency core cooling system (ECCS) room sump pumps were not required to be in service for ECCS and containment spray equipment operability. However, the USAR credits the ECCS sump pumps and their running lights for providing indication of the need for operator action in response to a passive failure in the ECCS system. The procedure changes were inappropriately categorized as being exempt from the 50.59 process because the changes involved "Managerial or Administrative Procedures." Because the issue affected the NRC's ability to perform its regulatory function, this finding was evaluated with the traditional enforcement process. The finding was determined to be of very low safety significance because, although the procedure changes could have resulted in the operation outside the requirements of technical specification action statements, no loss of function occurred. This was determined to be a Severity Level IV NCV of 10 CFR 50.59.
Inspection Report# : 2004015*(pdf)*

Significance: G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO COMPLY WITH TS BECAUSE OF DESIGN INADEQUACY IN THE STEAM FEEDWATER RUPTURE CONTROLS SYSTEM
A finding of very low safety significance was self-revealed when the licensee discovered, during planned work activities, that the Steam Feedwater Rupture Controls System logic cards could energize in a blocked condition after being de-energized. This condition could prevent automatic isolation of a faulted number 2 steam generator concurrent with a loss of offsite power. This condition was introduced into the system logic subsequent to a design change completed on the Steam Feedwater Rupture Controls System in 1988. When recognized in 2003, the licensee corrected the design deficiency. The inspectors concluded that the finding was greater than minor because it involved the attributes of design control and equipment reliability and could have affected the mitigating systems objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance because it did not result in an actual loss of safety function since the starting of the auxiliary feedwater system was not affected and a faulted number 2 steam generator could be isolated with operator action if automatic isolation did not occur. This was determined to be a Non-Cited Violation of Technical Specification 3.3.2.2.
Inspection Report# : 2004014*(pdf)*

Significance: G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO CORRECTLY VERIFY THE ADEQUACY OF THE DESIGN OF THE CONTAINMENT EMERGENCY SUMP SCREEN, AND TO IMPLEMENT EFFECTIVE CORRECTIVE ACTIONS TO REPAIR A 3/4-INCH WIDE BY 6-INCHES LONG GAP
A finding of very low safety significance was self-revealed during the licensee's inspection activities to address emergency core cooling system deficiencies, documented in LER 50-346/2002-005, and involved the licensee's failure to effectively implement corrective actions to verify the adequacy of the design of the containment emergency sump screen, and to implement effective corrective actions to repair an existing gap. Based on the inspectors' analysis, it was unlikely that the ECCS would be impacted. The inspectors concluded that the finding was greater than minor because the gap was associated with the objective and attributes of the Mitigating Systems and the Barrier Cornerstones. Specifically, the containment emergency sump screen was sized to pass no more than 1/4-inch debris particles and debris larger than 1/4-inch could have potentially damaged emergency core cooling system (ECCS) equipment and/or clogged the containment spray system (CSS) nozzles. The finding was of very low safety significance because: 1) the gap was not a design or qualification deficiency which resulted in a loss of function per Generic Letter 91-18, Revision 1; 2) did not represent an actual loss of safety function of a mitigating system; 3) did not represent an actual loss of safety function of a single train of a mitigating system for greater than its Technical Specification allowed outage time; 4) did not represent an actual loss of safety function of one or more non-Technical Specification mitigating system trains of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours; and 5) did not screen as potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event. In addition, containment spray is not a large early release frequency contributor per IMC 0609 Appendix H.

This was determined to be a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Actions.
Inspection Report# : 2004014*(pdf)*

Significance: SL-IV Sep 17, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Incomplete Information Provided to NRC in Licensing Submittal
The team identified that the licensee failed to provide complete information to the NRC in a licensing submittal. Specifically, the licensee did not identify that previously submitted licensing correspondence regarding the basis for not protecting ventilation system cables was no longer accurate.
Inspection Report# : 2004009*(pdf)*

Significance: SL-IV Jul 02, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Safety Evaluation for Changes to the Plant made as Described in the USAR Concerning the low-low pressure interlock for the AFW Pumps
The inspectors identified a Severity Level IV Non-Cited Violation associated with the failure to perform an adequate safety evaluation review as required by 10 CFR 50.59 for changes made to the facility as described in the Updated Safety Analysis Report. Specifically, the licensee failed to perform a safety evaluation in accordance with 10 CFR 50.59 for changes made to Section 9.2.7.3.c of the Updated Safety Analysis Report concerning the low-low pressure interlock for the auxiliary feedwater pumps. The changes made by the licensee adversely affected an Updated Safety Analysis Report-described function in that a previously described automatic feature of the steam inlet valve to the auxiliary feedwater pump was changed to clarify that this automatic feature was not available under certain conditions.

Because the Significance Determination Process is not designed to assess the significance of violations that potentially impact or impede the regulatory process, this issue was dispositioned using the traditional enforcement process in accordance with Section IV of the NRC Enforcement Policy. However, the results of the violation, that is, the failure to evaluate the changes made to Section 9.2.7.3.c of the USAR, were assessed using the Significance Determination Process.

This finding was determined to be more than minor because the inspectors could not determine reasonably that the change would not ultimately require NRC approval. The inspectors determined that this issue was of very low safety significance, because the design basis safety-related function of the auxiliary feedwater system, to remove reactor decay heat following a loss of normal feedwater, was not adversely affected, and because the team determined from the mitigating systems evaluation in the Phase 1 Screening Worksheet that all the questions were answered "No." Therefore, the results of the violation were determined to be of very low safety significance and the violation was classified as a Severity Level IV Violation. (Section 1R02)

Inspection Report# : 2004010*(pdf)*

Significance: SL-IV Jul 02, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate 10 CFR 50.59 Evaluation Regarding Tornado Missile Protection for EDG Exhaust Stacks
The inspectors identified a Severity Level IV Non-Cited Violation of 10 CFR 50.59, "Changes, Tests, and Experiments," based on the licensee performing an inadequate evaluation of a proposed change to the plant, regarding tornado missile protection of the diesel generator exhaust stacks and plant doors. Specifically, the licensee's response to the question posed in 10 CFR 50.59(c)(2)(vi) did not demonstrate that the proposed change did not create the possibility of a malfunction of equipment important to safety with a different result than any previously evaluated in the Final Safety Analysis Report (as updated).

Because the Significance Determination Process is not designed to assess the significance of violations that potentially impact or impede the regulatory process, this issue was dispositioned using the traditional enforcement process in accordance with Section IV of the NRC Enforcement Policy. However, the results of the violation, that is, the failure to demonstrate that the proposed change did not create the possibility of a malfunction of equipment important to safety with a different result, were assessed using the Significance Determination Process.

This finding was determined to be more than minor because the inspectors could not determine reasonably that the change would not ultimately require NRC approval. The finding was determined to be of very low safety significance based on a significance determination process analysis of a loss of offsite power concurrent with loss of one emergency diesel generator and the violation was classified as a Severity Level IV Violation. (Section 1R02)

Inspection Report# : 2004010*(pdf)*

Significance: G Jun 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO ADEQUATELY IMPLEMENT A PROCEDURE REQUIRED BY TS 6.8.1 THAT REQUIRED VERIFICATION

THAT CORRECT VALVE DATA WAS BEING USED TO TEST AND ADJUST MAIN STEAM SAFETY VALVES
A Non-Cited Violation of Technical Specification 6.8.1 was self-revealed when the licensee discovered that data, inputted to equipment being used to test and set the relief pressures of main steam safety valves, was incorrect. It was discovered that the incorrect valve parameters were being used for safety valves with setpoints of 1050 psig. The licensee failed to implement the procedurally specified data verification. The inspectors determined that the finding was more than minor because the licensee, by not effectively implementing an approved procedure for use of the test equipment, set a main steam safety valve outside of the acceptable range and declared the valve operable. The finding was of low safety significance because main steam relief capability remained sufficient and all activities causing entries into technical specification action statements were completed within technical specification allowable time limits.
Inspection Report# : 2004008*(pdf)*

Significance: G Jan 07, 2004
Identified By: NRC
Item Type: VIO Violation
Failure to Take Corrective Actions for a Previous NCV Concerning SW Pump Discharge Check Valve Acceptance Criteria
The team identified a Cited Violation of Technical Specifications Section 4.05a and 10 CFR 50.55a. Specifically, the licensee failed to ensure that the service water discharge check valve was tested in accordance with the American Society of Mechanical Engineers Code. The primary cause of this violation was related to the cross-cutting areas of problem identification and resolution and human performance, because the licensee did not recognize that the corrective actions taken needed to ensure compliance with NRC requirements.

The issue was determined to be more than minor because the inadequate test acceptance criteria allowed the licensee to accept a check valve as performing its intended function at less than full system flow. The issue was of very low safety significance using the Phase 1 of the significance determination process based on the licensee's determination that the system was operable but degraded. (Section 4OA3(3)b.12)

Inspection Report# : 2003010*(pdf)*

Significance: G Jan 07, 2004
Identified By: NRC
Item Type: VIO Violation
Failure to Take Corrective Actions for a Previous NCV Concerning SW Discharge Path Swapover Setpoints
The team identified a Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, the licensee failed to provide a basis for the setpoint to swap the service water system discharge path. This issue was previously identified as a Non-Cited Violation in Inspection Report 05000346/2002014 and the corrective actions taken by the licensee failed to correct the originally identified condition. The primary cause of this violation was related to the cross-cutting areas of problem identification and resolution and human performance, because the licensee did not recognize that the corrective actions taken needed to restore compliance with the identified violation of NRC requirements.

The issue was determined to be more than minor because the licensee had not corrected a previous violation and was relying on non-safety-related equipment to perform a safety function under design bases conditions. Because the issue was previously determined to be of very low safety significance, NRC management concluded that the violation could be categorized as having very low safety significance. (Section 4OA3 (3)b.11)
Inspection Report# : 2003010*(pdf)*

Significance: W Oct 08, 2003
Identified By: NRC
Item Type: VIO Violation
POTENTIAL INABILITY FOR HPI PUMPS TO PERFORM SAFETY RELATED FUNCTION
The failure of the licensee to correctly design the HPI pumps for accident mitigation during the recirculation mode of emergency core cooling.
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2004005*(pdf)*

Significance: Y Jun 30, 2003
Identified By: NRC
Item Type: VIO Violation
FAILURE TO EFFECTIVELY IMPLEMENT CORRECTIVE ACTIONS FOR DESIGN CONTROL ISSUES RELATED TO DEFICIENT CONTAINMENT COATINGS, UNCONTROLLED FIBROUS MATERIAL AND OTHER DEBRIS
An Apparent Violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to promptly identify and correct significant conditions adverse to quality regarding the implementation of corrective actions for design control issues related to deficient containment coatings, uncontrolled fibrous material and other debris. This impacted the ability of the emergency core cooling system sump to perform its function under certain accident scenarios due to clogging of the sump screen by unqualified coatings, fibrous materials, and various other debris. The issue is more than minor because the failure to implement appropriate corrective actions resulted in an actual loss of safety function of the ECCS system. The significance determination evaluation for this finding is documented in this report.

Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2003015*(pdf)*
Inspection Report# : 2004014*(pdf)*

Barrier Integrity

Significance: G Feb 19, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
LICENSEE FAILURE TO IDENTIFY LLRT CONFIGURATION NECESSARY TO ADEQUATELY PERFORM TYPE C LEAK TEST FOR NORMALLY OPEN MANUAL ISOLATION VALVES
A finding of very low safety significance was identified by the inspectors for a violation of Technical Specification Containment Leakage Rate Testing Program requirements. The licensee did not adequately perform Type C testing to identify potential leak paths through the bonnet and packing of normally locked opened manual isolation valves associated with containment penetrations for the containment spray system. The primary cause of this violation was related to the cross-cutting area of Human Performance. Licensee personnel, over several operational and test cycles, did not identify and appropriately test the potential leak paths although the licensee identified references in their leakage program, which stated that such testing was needed to comply with regulatory requirements. The issue was more than minor because, if left uncorrected, the issue could become a more significant safety concern because the potential containment leakage path was not being tested in accordance with the requirements for Type C testing. The issue was of very low safety significance because the leakage pathways, when tested, were within acceptable values. The issue was a Non-Cited Violation of Technical Specification 6.16 which required the establishment of a Containment Leakage Rate Testing program as required by 10CFR50.54(o) and 10CFR50, Appendix J.
Inspection Report# : 2005002*(pdf)*

Significance: G Jun 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO ADEQUATELY PERFORM PLANNED MAINTENANCE ACTIVITIES DEVELOPED TO ENSURE THAT PROTECTED BUILDING ROOF DRAINS AND OVERFLOW PIPES IN ROOF PARAPETS WERE NOT DAMAGED OR BLOCKED
A finding of very low safety significance was identified by the inspectors when they identified that the licensee failed to adequately perform planned maintenance activities developed to ensure that protected auxiliary building roof drains and overflow pipes in roof parapets were not damaged or blocked. The auxiliary building is a safety related structure. The inspectors determined that the finding was more than minor because, if left uncorrected, physical design barriers that protect the public from radionuclide releases caused by accidents or events could be challenged during a probable maximum precipitation event. The finding has very low safety significance since, in accordance with the Phase 1 Screening Worksheet of Inspection Manual Chapter 0609, "Significance Determination Process," the finding only represented a potential degradation of the radiological barrier function provided for the spent fuel pool. This issue was not an immediate safety concern, because, once identified by the inspector, the licensee took prompt action to clear the affected parapet drain screens. This issue was determined to be a a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V.
Inspection Report# : 2004008*(pdf)*

Significance: R Aug 09, 2002
Identified By: NRC
Item Type: FIN Finding
FAILURE TO PROPERLY IMPLEMENT THE BORIC ACID CONTROL AND THE CORRECTIVE ACTION PROGRAMS (EA 03-025)
The performance deficiency was the licensee's failure to properly implement the boric acid control and the corrective action programs, which allowed reactor coolant system pressure boundary leakage to occur undetected for a prolonged period of time resulting in reactor pressure vessel head degradation and control rod drive nozzle circumferential cracking.

The performance deficiency resulted in an increase in the risk of reactor core damage through a loss of coolant accident caused by either a rupture in the exposed cladding in the reactor pressure vessel head cavity or a control rod drive mechanism nozzle ejection due to a circumferential crack. The result of NRC's significance analysis of the as-found reactor pressure vessel head cavity and potential for larger cavity growth indicate that the significance is in the Red range (change in core damage frequency > 10-4 per reactor-year). The result of NRC's significance analysis of the as-found circumferential crack and potential for crack growth indicate that the significance is in the Yellow to Red range (change in core damage frequency in the range of low 10-5 to low 10-4 per reactor-year). Consequently, the NRC has determined that the performance deficiency resulting in the reactor pressure vessel head degradation and control rod drive mechanism nozzle cracking has high safety significance in the Red range.
Inspection Report# : 2003016*(pdf)*
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2005013*(pdf)*

Significance: N/A Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
I.A - Operating with Reactor Coolant Pressure Boundary Leakage

Technical Specification 3.4.6.2.a, Amendment 220, dated April 14, 1998, requires, in part, that the licensee shall limit reactor coolant system leakage to "No PRESSURE BOUNDARY LEAKAGE" during Modes 1 through 4.

Contrary to the above, between May 18, 2000, and February 16, 2002, the licensee started up and operated the plant in Modes 1 through 4 with reactor coolant system pressure boundary leakage, i.e. control rod drive penetration leakage. Specifically, the licensee returned the plant to operation following the 2000 refueling outages without fully characterizing and eliminating reactor coolant system pressure boundary leakage on the reactor pressure vessel head as evidenced by significant boric acid deposits on the reactor pressure vessel head at the start and end of the outage and by the development of new and extensive boric acid deposits on reactor containment equipment during the operation cycle.

This is a violation associated with a RED SDP finding. Civil Penalty - $5,000,000 (EA-05-071)
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2005013*(pdf)*

Significance: N/A Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
II.A 1 & 2 & 3 - Three examples of inadequate corrective actions involving (1) fouling of containmnet air coolers, (2) fourling of containment rad monitors, and (3) increased trend in unident. leakag

10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that the licensee shall establish measures to ensure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. Criterion XVI also requires that for significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and that corrective actions are taken to preclude repetition.

Plant Procedure NG-NA-00702, "Corrective Action Program," Revision 3, defined a significant condition adverse to quality to be a condition adverse to quality, which, if left uncorrected, could have an undesirable effect on plant safety, personal safety, regulatory position, financial liability, or environmental impact.

Contrary to the above, the licensee failed to determine the root cause of and take corrective actions to preclude the repetition of:

1. Fouling of containment air cooling fins by boric acid, between June 2000 and February 16, 2002, a significant condition adverse to quality.

2. - Fouling of the containment radiation elements by boric acid and iron oxide, between April 2001, and February 16, 2002, a significant condition adverse to quality, and

3. - An increasing trend in unidentified reactor coolant system leakage, between March 2001, and December 2001, a significant condition adverse to quality.

This is a violation associated with a RED SDP finding (EA-03-025).
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2005013*(pdf)*

Significance: N/A Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
II.B - Inadequate Boric Acid Corrosion Control procedure.

10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Procedure NG-EN-00324, "Boric Acid Corrosion Control Program," Revisions 0 through 2 (effective date October 1, 1999), were classified as a procedure affecting quality under the licensee's administrative system.

Contrary to the above, between October 1, 1999, and March 6, 2002, Procedure NG-EN-00324, "Boric Acid Corrosion Control Program," Revisions 0 through 2, were not appropriate to the circumstances and contributed to the licensee's failure to detect and address boric acid corrosion of the reactor vessel head, as follows:

1 - The procedure inappropriately focused on bolted and flanged connections in the definition of leakage (Sections 4.2 though 4.4), the definition of reactor coolant system pressure boundary components (Section 4.9), and the identification of investigation locations (Section 6.1) at the expense of identifying the potential for through-wall leakage.

2 - The procedure did not include adequate guidance, specifications, or threshold levels for initiating a "detailed inspection" in order to ensure consistent implementation of Section 6.3.4 of the procedure.

3 - The procedure did not require the identification of and corrective actions to preclude the repetition of boric acid leaks, a significant condition adverse to quality, but instead only required the preparation of a repair tag or work order to facilitate repair of the leak.

4 - The procedure did not define the qualifications and training necessary to permit engineering staff to conduct inspections and evaluations in a consistent manner, including the use of proper inspection techniques, observations, recording of results, and evaluations.

5 - The procedure inappropriately exempted stainless steel or Inconel components from further examination related to boric acid corrosion, unless the examination was during an ASME Section XI test which might require a bolting examination.

6 - The procedure inappropriately did not require the licensee staff to maintain records necessary to demonstrate the proper completion of activities affecting quality.

This is a violation associated with a RED SDP finding (EA-03-025).
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2005013*(pdf)*

Significance: SL-II Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
I.D - Failure to comply with Boric Acid Control procedure in that obstacles were not removed to allow for a complete inspection of the RPV head (failure to implement mod)
10 CFR 50, Appendix B, Criterion V, requires, in part, that activities affecting quality be accomplished in accordance with written procedures.

Davis-Besse Station Procedure NG-EN-00324, "Boric Acid Corrosion Control Program," Revisions 1/C1 and 2, Step 6.3.1, required, in part, that an initial inspection of boric acid buildup shall be performed to determine the "as found" conditions and to document the inspection results. The procedure also required, in Attachment 3, that insulation and other hindrances to direct visual [inspection] be removed as needed to allow detailed inspections of components suspected of leakage.

Potential Condition Adverse to Quality (PCAQ) 96-0551, initiated on April 21, 1996, documented the licensee's inability to comply with some inspections of the reactor pressure vessel head, as required by Procedure NG-EN-00324, and an inability to accurately determine the reactor pressure vessel head "as found" conditions, associated with boric acid deposits on the reactor pressure vessel head, due to the restrictions resulting from the location and size of the inspection ports, "mouse holes." The PCAQ further documented that only 50 to 60 percent of the reactor pressure vessel head could be inspected using the current inspection ports.

Modification 94-0025, initiated on May 27, 1994, and referenced as corrective action for PCAQ 96-0551, directed the completion of modifications to the reactor pressure vessel head service structure inspection ports to permit the inspection and cleaning of 100 percent of the reactor vessel head in accordance with Procedure NG-EN-00324.

Contrary to the above, on May 18, 2000, and at the end of Refueling Outage 12, the licensee failed to remove obstructions, including boric acid deposit buildups, necessary to conduct a detailed inspection of the reactor pressure vessel head and other components that may be suspected of leakage, as required by Plant Procedure NG-EN-00324, "Boric Acid Corrosion Control Program." The licensee's ability to conduct the inspections was significantly limited as a result of its concurrent deferral of the installation of Modification 94-0025, a corrective action for a significant condition adverse to quality documented in PCAQ 96-0551 and associated with the licensee's failure during previous outages to conduct complete inspections and cleaning of boric acid deposits on the reactor pressure vessel head.

This is a Severity Level II violation (Supplement I) Civil Penalty $110,000 (EA-05-067)
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2005013*(pdf)*

Significance: SL-I Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
I.B.1 & 2 - Information Included in (1) CR 2000-1037 and (2) WO 00-001846-000 was not complete and accurate in all material respects.
10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

10 CFR 50, Appendix B, Criterion XVI requires, in part, that for significant conditions adverse to quality, the cause of the condition and the corrective actions taken to preclude repetition shall be documented.

10 CFR 50, Appendix B, Criterion XVII, requires, in part, that the licensee shall maintain sufficient records to furnish evidence of activities affecting quality and that those records shall include monitoring of work performance.

Condition Report (CR) 2000-1037, closed May 1, 2000, documented corrective actions for the presence of boric acid on the reactor pressure vessel head, a significant condition adverse to quality, that included: "Accumulated boron deposited between the reactor head and the thermal insulation was removed during the cleaning process performed under W.O. [Work Order] 00-001846-000. No boric acid induced damage to the head surface was noted during the subsequent inspection."

Work Order 00-001846-000, "Clean Boron Accumulation from Top of Reactor Head and Top of Insulation," dated April 25, 2000, required the licensee staff to "clean boron accumulation from top of reactor head and on top of insulation." The Work Order Log, included as Page Four of the completed Work Order, documented that the, "work [was] performed without deviation" and was signed by the System Engineer on April 25, 2000.

Contrary to the above,

1 - The information included in CR 2000-1037 relative to the completed corrective actions and the subsequent inspection results were not complete and accurate in all material respects. Specifically, the licensee did not remove the accumulated boron deposits from all areas between the reactor head and the thermal insulation and did not conduct subsequent inspections of the entire reactor head. Instead, the licensee removed accumulated boric acid deposits from a portion of the reactor vessel head and conducted subsequent inspections for those portions of the reactor vessel head where the boric acid deposits had been removed.

2 - The Work Order Log, included as Page Four of completed Work Order 00-001846-000, a record required by Commission regulations to furnish evidence of activities affecting quality, contained information that was not accurate in all material respects. Specifically, the Work Order Log indicated that boron accumulation was cleaned from the top of the reactor head and on top of the insulation, without deviation, when, in fact, boric acid deposits were left on the head after the cleaning was completed on April 25, 2000.

This is a Severity Level I violation (Supplement VII). Civil Penalty $110,000 (EA-05-068)
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2005013*(pdf)*

Significance: SL-I Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
I.E1 & 2 - Licensee Responses to Bulletin 2001-01 dated (1) September 4, 2001, and (2) October 17, 2001, were materially incomplete and inaccurate.
10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," required all holders of operating licenses for pressurized water nuclear power reactors to provide information related to the structural integrity of the reactor vessel head penetration (VHP) nozzles for their respective facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements.

Contrary to the above, the licensee, a holder of an operating license for a pressurized water nuclear power reactor, the Davis-Besse Station, provided the Commission responses to Bulletin 2001-01 which included materially inaccurate and incomplete information as follows:

1 - In a September 4, 2001, response to the Bulletin entitled, "Response to Bulletin 2001-01," Serial 2731, the licensee made the following four materially inaccurate and incomplete statements:

(a) The licensee's response to Bulletin Item 1.c, on page 2 of 19, stated: "the minimum gap being at the dome center of the RPV [reactor pressure vessel] head where it is approximately 2 inches, and does not impede a qualified visual inspection."

The licensee's response was materially inaccurate, in that, the statement contradicted statements in the licensee's documents identified as PCAQR 94-0295 and 96-0551, which clearly stated that inspection capability at the top of the reactor vessel head was limited. The limitation was stated to be caused by the restricted access to the area through the service structure "weep holes", the curvature of the reactor pressure vessel head, and by the limited space to manipulate a camera due to the insulation that creates the two inch gap.

(b) The licensee's response to Bulletin Item 1.d, which requested inclusion of a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the reactor pressure vessel head for visual examinations, did not include a description of any limitations.

The licensee's response was materially incomplete in that the response did not mention that accessibility to the bare metal of the reactor pressure vessel head was impeded, during the Eleventh (1998) and the Twelfth (2000) Refueling Outages, by the presence of significant accumulations of boric acid deposits.

(c) The licensee's response to Bulletin Item 1.d, which also requested a discussion of the findings of reactor pressure vessel head inspections, stated that for the Twelfth Refueling Outage (2000), the inspection of the reactor pressure vessel head/nozzles indicated some accumulation of boric acid deposits.

The licensee's response was materially incomplete and inaccurate in that it mischaracterized the accumulation of boric acid on the reactor pressure vessel head and did not mention the evidence of corrosion that was evidenced by the pictures and the video examination of reactor pressure vessel head conditions documented at the beginning and ending of the Twelfth Refueling Outage (2000).

(d) The licensee's response to the Bulletin, on Page 3, stated: "The boric acid deposits were located beneath the leaking flanges with clear evidence of downward flow. No visible evidence of nozzle leakage was detected."

The licensee's response was materially inaccurate in that the boric acid deposits were not all located under leaking flanges and the licensee lacked clear evidence of the absence of downward flow for all nozzles. Specifically, the presence of boric acid deposits was not limited only to the areas beneath the flanges, as implied by that statement. The build-up of boric acid deposits was so significant that the licensee could not inspect all of the nozzles. As a result, the licensee also did not have a basis for stating that no visible evidence of nozzle leakage was detected.

2. In an October 17, 2001, response to the Bulletin entitled, "Supplemental Response to Bulletin 2001-01," Serial 2735, the licensee stated: "In May 1996, during a refueling outage, the RPV [reactor pressure vessel] head was inspected. No leakage was identified, and these results have been recently verified by a re-review of the video tapes obtained from that inspection. The RPV head was mechanically cleaned at the end of the outage. Subsequent inspections of the RPV head in the next two refueling outages (1998 and 2000), also did not identify any leakage in the CRDM [control rod drive mechanism] nozzle-to-head areas that could be inspected. Video tapes taken during these inspections have also been re-reviewed."

The licensee's response was materially inaccurate, in that: (1) each reactor pressure vessel head control rod drive penetration was not inspected in May 1996, as documented in PCAQR 96-0551, and; (2) the reactor pressure vessel head, including the area around each control rod drive penetration, was not completely cleaned, as noted in PCAQR 98-0649, which was prepared at the start of the Eleventh Refueling Outage (1998), which stated that there were old boric acid deposits on the head.

This is a Severity Level I violation (Supplement VII) Civil Penalty $120,000 (EA-05-072)
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2005013*(pdf)*

Significance: SL-II Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
I.C. 1 & 2 & 3 - Failure to determine the cause of a signficant condition adverse to quality involving three examples of identified boric acid leakage.
10 CFR 50, Appendix B, Criterion XVI, requires, in part, that licensees shall establish measures to ensure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. For significant conditions adverse to quality, the licensee shall establish measures to ensure that the cause of the condition is determined and that corrective actions are taken to preclude repetition.

Plant Procedure NG-NA-00702, "Corrective Action Program," Revision 3, defined a significant condition adverse to quality to be a condition, which, if left uncorrected, could have an undesirable effect on plant safety, personal safety, regulatory position, financial liability, or environmental impact.

Contrary to the above, the licensee did not determine the cause of the condition and did not implement corrective actions to preclude repetition of the condition associated with the identification and removal of boric acid on the reactor vessel head, a significant condition adverse to quality, prior to closing the associated condition reports.

Specifically:

1 - On April 27, 2000, the licensee closed CR 2000-0781, "Leakage from CRD [Control Rod Drive] Structure Blocked Visual Exam of Reactor Vessel Head Studs," issued on April 6, 2000, associated with the accumulation of boric acid deposits on the reactor vessel head studs without determining the cause of the deposits, i.e., identifying the source of the reactor coolant system leakage, and without taking corrective actions to preclude recurrence.

2 - On April 27, 2000, the licensee closed CR 2000-0782, "Inspection of Reactor Flange Indicated Boric Acid Leakage From Weep Holes," issued on April 6, 2000, associated with the accumulation of boric acid deposits on the reactor vessel head, without determining the cause of the boric acid deposits, i.e., identifying the source of the reactor coolant system leakage, without removing all of the known boric acid deposits on the reactor pressure vessel head, and without taking corrective actions to prevent recurrence.

3 - On May 1, 2000, the licensee closed CR 2000-1037, "Inspection of Reactor Head Indicated Accumulation of Boron in Area of the CRD [Control Rod Drive] Nozzle Penetration," issued on April 17, 2000, associated with the accumulation of boric acid deposits on the reactor vessel head, without determining the cause of the boric acid deposits, i.e., identifying the source of the reactor coolant system leakage, without removing all of the known boric acid deposits on the reactor vessel head, and without taking corrective actions to prevent recurrence.

This is a Severity Level II violation (Supplement I) Civil Penalty - $110,000 (EA-05-066)
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2005013*(pdf)*

Significance: SL-IV Sep 04, 2001
Identified By: NRC

Item Type: VIO Violation
II.D 1 & 2- Two examples of incomplete and ainaccurate information contained in quality documents, (1) a void request dated 9/12/93, and (2) a QA Audit Report No. AR-00-OUTAG-01.
10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

10 CFR Part 50, Appendix B, Criterion XVII, requires, in part, that the licensee shall maintain sufficient records to furnish evidence of activities affecting quality and that those records shall include audits and those actions taken to correct any deficient conditions.

Contrary to the above, the following information was not complete or accurate in all material respects for documents required to be maintained or provided to the Commission:

1 - On September 23, 1993, the licensee processed a "Document Void Request" to cancel Modification 90-012 which stated, "Current inspection techniques using high-powered cameras preclude the need for inspection ports, additionally, cleaning of the reactor vessel head during last three outages was completed successfully without requiring access ports." However, the quoted statement was not accurate in all material respects, in that, the licensee left boric acid deposits on the reactor vessel head at the end of both the seventh and eighth refueling outages, the two outages preceding this statement.

2 - Quality Assurance Audit Report AR-00-OUTAG-01, dated July 7, 2000, stated, in part, "Boric Acid Corrosion Control Checklists and Condition Reports were initiated by inspectors when prudent to document and evaluate boric acid accumulation and leaks. Boric acid leakage was adequately classified and corrected when appropriate. Engineering displayed noteworthy persistence in ensuring boric acid accumulation from the reactor head was thoroughly cleaned." However, the audit report was not accurate in all material respects in that the licensee did not: 1) thoroughly clean the reactor head during the outage; 2) did not prepare a boric acid corrosion control checklist for the boric acid left on the head after the cleaning attempt; and 3) identify, properly classify, or correct the boric acid accumulation and leaks.

This is a Severity Level IV violation (Supplement VII) (EA-05-070)
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2005013*(pdf)*

Significance: SL-III Apr 25, 2000
Identified By: NRC
Item Type: VIO Violation
II.C - Two PCAQs (98-0649 & 98-0767) were closed as completed based on inaccurate informationon.
10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

10 CFR Part 50, Appendix B, Criterion XVII, requires, in part, that the licensee shall maintain sufficient records to furnish evidence of activities affecting quality and that those records shall include actions taken to correct any deficient conditions.

Contrary to the above, the following information was not complete or accurate in all material respects for documents required to be maintained or provided to the Commission:

1 - Potential Condition Adverse to Quality Report (PCAQR) 98-0649, dated April 18, 1998, contained the following closure statement: "Accumulation of boric acid on the reactor vessel caused by leaking CRDMs [control rod drive mechanisms] has not resulted in any boric acid corrosion. This was identified through inspections following reactor vessel head cleaning in past outages....Additionally, B&W [Babcock & Wilcox] documentation discussing CRDM nozzle cracking further stated that boric acid deposits on the head caused by leaking CRDM flanges would not result in head corrosion." However, the quoted statements were not accurate in all material respects in that the licensee had previously not cleaned all areas of the reactor head of boric acid deposits, had not inspected the base metal under all the deposits to determine whether corrosion was present, and no B&W documentation was available to support the claim that boric acid would not result in head corrosion.

2 - Potential Condition Adverse to Quality Report (PCAQR) 98-0767, dated April 25, 1998, Section 4A, Item F, included the following closure justification, "The boric acid deposits were removed from the head." However, the quoted statement was not accurate in all material respects in that the licensee had not removed all of the boric acid deposits from the head as of the end of the eleventh refueling outage.

This is a Severity Level III violation (Supplement VII) (EA-05-069)
Inspection Report# : 2005012*(pdf)*
Inspection Report# : 2005013*(pdf)*

Emergency Preparedness

Significance: W Nov 29, 2004
Identified By: NRC

Item Type: VIO Violation
Final White Finding - Ottawa County Officials Lost Capability to Activate All EPZ Sirens for About 10 Days & Had Degraded Capability for About 30 Days
The Ottawa County Sheriff Dispatch Center lost its capability to activate all 54 EPZ sirens from April 27 through May 7, 2004, (10 days) and its capability to activate all 54 EPZ sirens was degraded from April 6 through May 7, 2004 (30 days).

The finding was more than minor because if Ottawa County officials needed to promptly activate the EPZ sirens between the morning of April 27 and noontime on May 7, 2004, the EPZ sirens would not have activated and the safety of some of the EPZ population could have been adversely impacted. The preliminary White finding was associated with one of the attributes of the Emergency Preparedness Cornerstone, specifically the facilities and equipment attribute. This preliminary finding was assessed using the Emergency Preparedness SDP and was considered to be a degradation of the risk significant emergency planning standard 10 CFR 50.47(b)(5).

The NRC determined that the final significance of this item was White.
Inspection Report# : 2004018*(pdf)*

Significance: SL-IV Nov 29, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Submital of Discrepant ANS PI Data for Second & Third Quarters of 2004
Inspectors identified that the ANS PI data submitted by the licensee for the second and third calendar quarters of 2004 were discrepant. The licensee incorrectly interpreted Revision 2 of the NEI 99-02 that provided criteria on what ANS tests could be counted as PI opportunities. The licensee incorrectly counted its "silent" ANS tests beginning on June 1, 2004 through September 30, 2004. As a result of counting "silent" tests, the reported ANS PI data remained in the GREEN band throughout the 12 month periods ending on June 30 and September 30, 2004. In contrast, NRC concluded that the ANS PI should have been in the WHITE band at the ends of both 12 month periods.
Inspection Report# : 2004018*(pdf)*
Inspection Report# : 2005010*(pdf)*

Significance: W Nov 29, 2004
Identified By: NRC
Item Type: VIO Violation
Final White Finding - Ottawa County Officials Lost Capability to Activate All EPZ Sirens for About 10 Days & Had Degraded Capability for About 30 Days
The Ottawa County Sheriff Dispatch Center lost its capability to activate all 54 EPZ sirens from April 27 through May 7, 2004, (10 days) and its capability to activate all 54 EPZ sirens was degraded from April 6 through May 7, 2004 (30 days).

The finding was more than minor because if Ottawa County officials needed to promptly activate the EPZ sirens between the morning of April 27 and noontime on May 7, 2004, the EPZ sirens would not have activated and the safety of some of the EPZ population could have been adversely impacted. The preliminary White finding was associated with one of the attributes of the Emergency Preparedness Cornerstone, specifically the facilities and equipment attribute. This preliminary finding was assessed using the Emergency Preparedness SDP and was considered to be a degradation of the risk significant emergency planning standard 10 CFR 50.47(b)(5).

The NRC determined that the final significance of this item was White.
Inspection Report# : 2005010*(pdf)*

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Significance: G Dec 16, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO FOLLOW CORRECTIVE ACTION PROCEDURE & IMPLEMENT REMEDIATION REQUIREMENTS FOLLOWING REJECTION OF TWO APPARENT CAUSE EVALUATIONS
The inspectors identified a finding of very low significance associated with a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings, for failure to follow the corrective action procedure and implement remediation requirements following rejection of two apparent cause evaluations. This inspector-identified issue was greater than minor because if left uncorrected the issue would become a more significant safety concern involving the implementation of the corrective action program for significant conditions adverse to quality and conditions adverse to quality. The inspectors determined that the finding was not suitable for SDP evaluation because the failure to remediate the evaluators of the condition reports did not directly result in degraded equipment. Therefore, this finding was reviewed by Regional Management, in accordance with IMC 0612 Section 05.04c, and determined to be of very low safety significance. The finding was not greater than very low safety significance because no safety-related equipment was rendered inoperable.
Inspection Report# : 2004017*(pdf)*

Significance: G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO MEET PROCEDURE IMPLEMENTATION REQUIREMENTS OF 10 CFR 50, APPENDIX B, CRITERION V, ASSOCIATED WITH RECURRING OPERATIONS PERFORMANCE ISSUES
The inspectors identified a finding having very low safety significance regarding the licensee's failure to identify proper corrective actions to preclude repetition of conditions adverse to quality as required by the Corrective Action Program. Specifically, corrective actions implemented to address repetitive Technical Specification violations and licensed operator performance errors, were not effective in precluding recurrence of similar events. The finding was more than minor because, if left uncorrected, the issue would become a more significant safety concern. Because this finding did not directly affect any of the cornerstone attributes, it was reviewed by Regional Management, in accordance with IMC 0612 Section 05.04c. The finding was determined to be of very low safety significance because no safety systems were degraded nor was any safety equipment rendered inoperable directly due to this issue. The issue was a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings" which requires that activities affecting quality shall be accomplished in accordance with prescribed instructions and procedures.
Inspection Report# : 2004014*(pdf)*

Last modified : June 29, 2005