Byron 2
3Q/2014 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2014
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
FAILURE TO PROPERLY IMPLEMENT A COMPENSATORY FIRE WATCH AS REQUIRED BY THE FIRE PROTECTION PROGRAM
A finding with two examples of very low safety significance and associated NCV of Technical Specification 5.4.1.c was self-revealed when required compensatory fire watches were discovered to have been terminated while the fire systems were still impaired. Specifically, the licensee failed to maintain compensatory fire watches for Fire Zone 3.1-1, “Unit 1 Electrical Cable Tunnel” and for Fire Zones 10.1-2 “2B Diesel Fuel Oil Storage Room” and 10.2-2 “2A Diesel Fuel Oil Storage Room” required by procedure OP-MW-201-007 and as described in Technical Requirements Manual limiting conditions for operations.
The inspectors determined that this finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Screening,” dated September 7, 2012, because the finding was associated with the Initiating Events Cornerstone attribute of Protection Against External Factors (Fire) and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during plant operations. Specifically, required fire watches established as compensatory measures should have been maintained for the duration of the work activity so that the sites ability to promptly detect and suppress a fire would be maintained. The inspectors evaluated this issue in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Initial Characterization of Findings.” In Table 3 of Attachment 4, “SDP Appendix Router,” the inspectors answered “Yes” to Question E.2, “Does the finding involve:…(2) Fixed fire protection systems….?” Therefore, the inspectors continued the risk evaluation using IMC 0609 Appendix F, “Fire Protection Significance Determination Process.” Due to the equipment located in each of the affected fire zones, the two examples were evaluated independently. One example screened to Green using the questions under Task 1.4.2 for fixed fire protections systems. The senior reactor analyst performed a quantitative Phase 2 evaluation and determined the issue to be Green. The inspectors determined that a principle contributor to the finding was that the organization did not implement a process for planning, implementing, and executing concurrent work activities that ensured the required compensatory actions were maintained such that nuclear safety was the overriding priority (WP.1). As a result, the inspectors assigned a cross-cutting aspect of Work Management (H.5) to the finding.
Inspection Report# : 2014002 *(pdf)*

Significance: G Dec 31, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
Reactor Vessel Design Documents Not Updated to Reflect Unit 2 Missing Head Stud
Inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion III “Design Control” for failure to maintain reactor vessel design specification and analysis up-to-date for the 53 stud vessel head configuration applicable to Unit 2. Specifically, the reactor vessel Design Specification and Design Analysis did not reflect the modified and stuck stud no. 11. The licensee entered this issue into their corrective action program (CAP) as Action Report (AR) 01578285.
The inspectors determined that the performance deficiency was more than minor because it was associated with the

Design Control attribute of the Initiating Events Cornerstone and adversely impacted the cornerstone objective of to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The inspectors also answered “Yes” to the More-than-Minor screening question “If left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?” Specifically, the inspectors determined that this issue was more than minor because, if left uncorrected, the failure to maintain the Unit 2 reactor vessel design specification and analysis caused them to be inaccurate and if these documents were subsequently relied on for future design changes, the vessel design may not be adequate to maintain structural integrity during design basis events resulting in a loss-of-coolant-accident. The inspectors performed a Phase 1 Significance Determination Process Screening, and evaluated this issue by application of questions 1 and 2. Questions No’s 1 and 2 asked: if after a reasonable assessment of degradation, could the finding result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident (LOCA) or could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function (e.g., Interfacing System LOCA)? In this case, the degradation prompting the reduction in the number of head studs and the licensee’s failure to maintain the design analysis had not yet affected the ability of the reactor vessel to perform its design functions so the inspector answered these questions “No” and this issue screened as having very low risk significance. Inspectors determined that this finding is not indicative of current performance and no cross-cutting aspect was assigned.

Inspection Report# : 2013005 *(pdf)*

Significance: G Dec 31, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
Corrosion Effects on the Unit 2 Reactor Vessel Not Monitored
Inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion III “Design Control” for failure to maintain reactor vessel design specification and analysis up-to-date for the 53 stud vessel head configuration applicable to Unit 2. Specifically, the reactor vessel Design Specification and Design Analysis did not reflect the modified and stuck stud no. 11. The licensee entered this issue into their corrective action program (CAP) as Action Report (AR) 01578285.
The inspectors determined that the performance deficiency was more than minor because it was associated with the Design Control attribute of the Initiating Events Cornerstone and adversely impacted the cornerstone objective of to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The inspectors also answered “Yes” to the More-than-Minor screening question “If left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?” Specifically, the inspectors determined that this issue was more than minor because, if left uncorrected, the failure to maintain the Unit 2 reactor vessel design specification and analysis caused them to be inaccurate and if these documents were subsequently relied on for future design changes, the vessel design may not be adequate to maintain structural integrity during design basis events resulting in a loss-of-coolant-accident. The inspectors performed a Phase 1 Significance Determination Process Screening, and evaluated this issue by application of questions 1 and 2. Questions No’s 1 and 2 asked: if after a reasonable assessment of degradation, could the finding result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident (LOCA) or could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function (e.g., Interfacing System LOCA)? In this case, the degradation prompting the reduction in the number of head studs and the licensee’s failure to maintain the design analysis had not yet affected the ability of the reactor vessel to perform its design functions so the inspector answered these questions “No” and this issue screened as having very low risk significance. Inspectors determined that this finding is not indicative of current performance and no cross-cutting aspect was assigned.

Inspection Report# : 2013005 *(pdf)*

Significance: G Dec 31, 2013

Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Vessel Stress Analysis for Unit 2 Missing Head Stud
Inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion III "Design Control" for failure to perform an adequate thermal-mechanical analysis to support operation with a missing Unit 2 head stud. Specifically, the licensee did not perform a complete set of analysis under operating, faulted and design conditions to confirm the associated stud and flange stresses remained within the Code allowable limits. Consequently, the licensee did not recognize that the bearing stress under the head stud nuts at the vessel flange face exceeded the Code allowable stress. The licensee entered this issue into their CAP as IR 01578717.
The inspectors determined that the performance deficiency was more than minor because it was associated with the Design Control attribute of the Initiating Events Cornerstone and adversely impacted the cornerstone objective of to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The inspectors also answered "Yes" to the More-than-Minor screening question "If left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, the inspectors determined that this issue was more than minor because, if left uncorrected, the failure to perform an adequate thermal-mechanical analysis, could result in the inability of the reactor vessel to meet the design basis operating transient without a LOCA. The inspectors performed a Phase 1 Significance Determination Process Screening and evaluated this issue by application of questions 1 and 2. Questions 1 and 2 asked if; after a reasonable assessment of degradation, could the finding result in exceeding the reactor coolant leak rate for a small loss-of-coolant-accident (LOCA) or could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function (e.g., Interfacing System LOCA)? In this case, because of the available margins in the flange material strength, the inspector answered these questions "No" and this issue screened as having very low risk significance. This finding has a cross-cutting aspect in the area of Human Performance Resources, because the licensee did not have complete, accurate and up-to-date design documentation, procedures, and work packages. Specifically, the licensee failed to ensure the applicable ASME Code Section III design limit for bearing stress (design basis) was correctly translated into a design document (EC 379850). (H.2(c)).

Inspection Report# : 2013005 *(pdf)*

Mitigating Systems

Significance: G Dec 31, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
Emergency Service Water Blowdown Isolation Valves Were Not Tested
The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for failure to demonstrate the ability to isolate the emergency service water blowdown as credited in analysis described in the Updated Final Safety Analysis Report. Specifically, the licensee was not periodically testing the active function of the blowdown isolation valves. This finding was entered into the licensee's Corrective Action Program, in part, to periodically test the closing function of the blowdown isolation valves.
The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of the ultimate heat sink to respond to initiating events to prevent undesirable consequences. The finding screened as of very low safety significance because it did not result in the loss of operability or functionality. Specifically, the licensee reviewed recent history of the affected piping system and determined it opportunistically cycled the valves without incidents. The inspectors did not identify a cross-cutting

aspect associated with this finding because it was not confirmed to reflect current performance due to the age of the performance deficiency.

Inspection Report# : 2013005 *(pdf)*

Significance: G Dec 31, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
Intake Structure Silt Level Acceptance Criteria Were Non-Conservative
The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for failure to develop appropriate intake structure silt level acceptance criteria. Specifically, the licensee used a non-conservative river water low level value as an input when developing silt level acceptance criteria. This finding was entered into the licensee’s CAP to correct the acceptance criteria and revise the affected surveillance procedures.
The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of the ultimate heat sink to respond to initiating events to prevent undesirable consequences. The finding screened as of very low safety significance because it did not result in the loss of operability or functionality. Specifically, a historic review did not find an example where the as-found silt level resulted in an inoperable condition. The inspectors did not identify a cross-cutting aspect associated with this finding because it was not confirmed to reflect current performance due to the age of the performance deficiency.

Inspection Report# : 2013005 *(pdf)*

Significance: G Dec 31, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Implement Preventative Maintenance Procedure Repalacement Schedules for Essential Service Water Makeup Pump Diesel Engine Hoses
. Inspectors identified a finding of very low safety significance and associated Non-Cited Violation of TS 5.4.1, “Procedures,” for failure to establish and implement a preventive maintenance schedule to replace hoses on SX Make Up pump diesel engine. Specifically, the licensee failed to implement preventive maintenance procedures that require periodic replacement of hoses on pre-established schedules in accordance with vendor recommendation and corporate Performance Centered Maintenance (PCM) template. The finding was entered into the licensee’s Corrective Actions Program, in part, to evaluate the current maintenance strategy for maintaining flexible hoses on the SX make-up pump diesel engines.
The performance deficiency was determined to be more than minor because if left uncorrected the failure of SX Make up pump engine hoses could result in the in the inoperability of the SX Make up pumps. The performance deficiency also screened as more than minor because it affected the Procedure Quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as being of very low safety significance because it did not result in the loss of operability or functionality. Specifically, the licensee has reviewed the recent history of hose inspections and instances that required hose replacement and determined no failures have occurred that resulted in an inoperable condition. The inspectors did not identify a cross-cutting aspect associated with this finding because it was not confirmed to reflect current performance due to the age of the performance deficiency.
Inspection Report# : 2013005 *(pdf)*

Barrier Integrity

Significance: G Dec 31, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
Analytical Bases for PTL Curves Not Maintained Consitent With Unit 2 Head Stud Configuration
Inspectors identified a finding of very low safety significance and an associated NCV of TS 5.6.6, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), for failure to maintain the analytical basis for deriving the pressure temperature limit curves consistent with the Unit 2 vessel head stud configuration. Specifically, the analytical model used in WCAP-16143 "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2" was based the original closure head configuration and did not represent the modified closure head configuration (53 head studs) applicable to the Unit 2 reactor vessel. The licensee entered this issue into their CAP as AR 01578276.
The inspectors determined that the performance deficiency was more than minor because it was associated with the Design Control attribute of the Barrier Integrity Cornerstone and adversely impacted the cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accident or events. The inspectors also answered "yes" to the More-than-Minor screening question "If left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, if left uncorrected, continued operation without a correct stress analysis to support the approved pressure temperature limit curves could have allowed the reactor to operate at a pressure and temperature that increased the chance for a brittle fracture of the vessel under a pressurized thermal shock event. The inspectors performed a Phase 1 Significance Determination Process Screening and selected the box under the Reactor Coolant System Boundary (e.g. pressurized thermal shock issues) which required a detailed risk evaluation. An NRC Region III senior reactor analyst performed a detailed risk evaluation of this finding. A potential increase in the probability for vessel failure would exist if the plant was operated in the unacceptable pressure-temperature regions and a pressurized thermal shock event occurred. Based on the licensee and supporting vendor assessments which concluded that no substantial increase in vessel stresses will occur due to operation with 53 head studs, the driving force for crack propagation (e.g. K1) will remain essentially unchanged. However, to bound the delta risk evaluation, it was assumed that the initiating event frequency for a reactor vessel failure increased by 10 percent. From the Byron Standardized Plant Analysis Risk model version 8.21, the initiating event frequency for reactor vessel failure from any cause was 1E-7/yr. Core damage is expected to occur if reactor vessel failure occurs. The exposure time for the finding was the maximum of one year. Thus, a bounding risk assessment yields a delta risk of 1E 8/yr. Therefore, based on the detailed risk evaluation, this finding is of very low risk significance. This finding has a cross-cutting aspect in the area of Human Performance Decision Making, because the licensee did not use conservative assumptions in decision making and adopt a requirement to demonstrate that the proposed action was safe in order to proceed. In this case, the licensee staff made a non-conservative assumption that the 10 CFR 50.59 process could be applied to authorize a change in the WCAP 16143 analysis and to not seek prior NRC approval (H.1 (b)).

Inspection Report# : 2013005 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Last modified : November 26, 2014