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United States Nuclear Regulatory Commission
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Byron Station, Unit 1 and 2
Renewed Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: 10 CFR 50.59 Summary Report

Pursuant to the requirements of 10 CFR 50.59, "Changes, tests and experiments," paragraph (d)(2), Byron Station is providing the required report for Facility Operating License Nos. NPF-37 and NPF-66. This report is provided for the evaluations implemented for the time period of January 1, 2016 through November 30, 2018 and consists of 10 CFR 50.59 Review Coversheets for changes to the facility or procedures as described in the Updated Final Analysis Report (UFSAR) and test or experiments not described in the UFSAR.

Please direct any questions regarding this submittal to Mr. Douglas Spitzer, Regulatory Assurance Manager, at (815) 406-2800.

Respectfully,

Mark E. Kanavos
Site Vice President
Byron Generating Station

MEK/GC/rm

Attachment: Byron Station 10 CFR 50.59 Summary Report

cc: NRC Regional Administrator – NRC Region III

ATTACHMENT

Byron Station 10 CFR 50.59 Summary Report

#	Evaluation No.	Rev	Title
1	6G-15-001	0	ECCS/CS Hydraulic Analysis Incorporate Instrument Uncertainty from CS Pump Testing
2	6G-16-004	0	Replace RCP Under frequency KF Relays with Circuit Shield Type 81 Frequency Relays to Address ABB Part 21 Notification of Potential Defect For KF Relay ZPA (Unit 1)
3	6G-16-005	0	0BOA PRI-7 Loss of Ultimate Heat Sink Unit 0 (Revision 2)
4	6G-16-007	0	Temporarily Defeat FW Water Hammer Prevention System (WHPS) FW Isolation Signals During Normal Power Operation for Steam Generators 2A/2B/2C/2D
5	6G-16-008	0	APPENDIX J Scope Reduction as a Result of True North Engineering Report Commissioned by Corporate Engineering
6	6G-17-001	0	Lost Spent Fuel Pool (SFP) Crimps and RVLIS Pins
7	6G-17-002	0	Westinghouse Ovation Digital Upgrade for Rod Control Logic Cabinet (N-1 Outage)
8	6G-17-003	0	Westinghouse Ovation Digital Upgrade for DEHC
9	6G-17-004	0	TCCP to Install Pneumatic Jumper to Open 1WG074
10	6G-17-005	0	Provide Capability to Disable Water Hammer Prevention System (WHPS) Feedwater (FW) Isolation Signals During Normal Power Operation & Change WHPS SG Low Pressure/Level Control Relay Operation from Actuate On De-Energized To Actuate On Energized
11	6G-17-006	0	Process TORMIS LAR & Supporting Docs, TRM 3.7.e "Tornado Design Basis SXCT Fans - Operating", TRM 3.7.f "Tornado Design Basis SXCT Fans - Shutdown".
12	6G-17-007	0	Lost Parts Evaluation for Head Lift Rigging

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Units 1 and 2

Activity/Document Number: EC 392420 and DRP 17-034 **Revision Number:** 0/0

Title: ECCS/CS Hydraulic Analyses Incorporate Instrument Uncertainty from CS Pump Testing

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)
EC 392420 implements Revisions 4 and 5A of design analysis BYR06-029/BRW-06-0016-M. This analysis calculates flow rates for the Emergency Core Cooling System (ECCS) and Containment Spray (CS) System in support of the Braidwood and Byron Station design basis and accident analyses.

EC 392420 also includes design basis analyses that have been revised to incorporate the new ECCS and CS flow rates. The complete list of analyses that have been revised is as follows:

1. **BYR06-029/BRW-06-0016-M, "SI-RHR-CS-CV System Hydraulic Analysis in Support of GSI-191", Revision 4 and Revision 5A**
2. **ATD-0111, Maximum Containment Flood Level", Revision 15**
3. **SI-90-01 Revision, "Minimum Containment Flood Level", Revision 11**
4. **BRW-05-063-M/BYR05-061, "GSI-191 Evaluation of Long Term Downstream Effects", Revision 4**
5. **BRW-06-0035-M/BYR06-058, "NPSHA for RHR and CS Pumps During Post-LOCA Recirculation", Revision 3**
6. **BRW-97-0337-M/BYR97-191, "Containment Spray System Hydraulic Model", Revision 3**
7. **BRW-06-0028-M/BYR06-030, "Post-LOCA Chemical Effects Analysis in Support of GSI-191", Revision 3**
8. **BYR97-406/BRW-97-0965-M, "RCFC Performance Curve Calculation", Revision 1**
9. **SITH-1, "Refueling Water Storage Tank (RWST) Level Setpoints", Revision 8**
10. **CAE-07-49/CCE-07-48, "Phase 2 Evaluation of Reduced SI Flow During Recirculation Phase of ECCS", Revision 1A.**
11. **CN-CRA-07-47, "Byron/Braidwood Unit 1 Steam Generator Tube Rupture Margin to Overfill", Revision 4**
12. **CN-CRA-07-72, "Byron/Braidwood Unit 2 Steam Generator Tube Rupture Margin to Overfill", Revision 3**
13. **CN-CRA-09-67, "Byron/Braidwood Unit 2 Steam Generator Tube Rupture Input to Dose", Revision 1**
14. **CN-CRA-10-29, "Byron and Braidwood Units 1 and 2 Main Steamline Break IC/OC Evaluation of M&E Releases and Containment/Compartment Responses", Revision 1**
15. **CN-CRA-10-30, "Byron/Braidwood Unit 1 Steam Generator Tube Rupture Input to Dose Analysis", Revision 1**
16. **CN-CRA-10-54, "LOCA Mass and Energy Release and Containment Integrity Analysis", Revision 2**
17. **CN-CRA-13-5, "Byron and Braidwood Units 1: Steamline Break Mass and Energy Releases Outside Containment and Compartment Response", Revision 1**
18. **CN-CRA-13-6, "Byron and Braidwood Units 2: Steamline Break Mass and Energy Releases Outside Containment and Compartment Response", Revision 1**
19. **CN-CRA-13-21, "Byron and Braidwood Unit 2 – Steamline Break Mass and Energy Releases Inside Containment", Revision 0**
20. **CN-CRA-13-26, "Byron and Braidwood Unit 2 – Steamline Break Containment Response Analysis", Revision 1**
21. **CN-CRA-13-29, "Byron and Braidwood Unit 1 – Steamline Break Mass and Energy Releases Inside Containment", Revision 0**

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Units 1 and 2

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Title: ECCS/CS Hydraulic Analyses Incorporate Instrument Uncertainty from CS Pump Testing

22. **CN-CRA-14-10, "Byron and Braidwood Unit 1 – Steamline Break Containment Response Analysis", Revision 1**
23. **PSA-B-98-08, "Byron/Braidwood ECCS Flow Calculations for Safety Analysis", Revision 4**
24. **LTR-PL-13-32, "Byron/Braidwood Unit 1 and 2 Increased Delay Time in Auxiliary Feedwater Injection Final Engineering Report", Revision 001A**
25. **3 SA-096.018, "Head Loss Calculation", Revision 9**

Note: The common Byron/Braidwood accident analyses were revised in conjunction with a Braidwood change to the maximum UHS supply water temperature (Braidwood EC 396478). These analyses are conservative for Byron as they use inputs that are more limiting than the Byron maximum UHS temperature of 100 °F, described in UFSAR Section 9.2.1.2.1.

In addition, the impact on the Peak Clad Temperature (PCT) due to the revised ECCS flow rates has been evaluated to be an increase of 2 °F for the LBLOCA and 0 °F for the SBLOCA. The evaluations are documented in the following Westinghouse documents:

1. **Letter LTR-LIS-14-146, dated May 2, 2014**
2. **Letter LTR-LIS-14-237, dated October 22, 2014**
3. **Letter LTR-LIS-14-245, dated October 27, 2014**

The PCT evaluations and results were incorporated into the 10CFR50.46 Annual Report (Reference EC 401531).

DRP No. 17-034 incorporates the revised design analyses results into the UFSAR for Byron. UFSAR Chapter 6.2 is revised to document the inputs and results of the revised analyses of containment response to postulated line breaks inside containment. The UFSAR Chapter 6.3 description of the minimum CS pump flow/CS spray nozzle flow is revised and the RH, CV, and SI pump performance curves are revised.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

Revision 4 to design analysis BYR06-029/BRW-06-0016-M incorporates instrument uncertainty applicable to the in-service testing for the CS pumps into the analysis. This is an item that was identified in Braidwood IRs 1048015, 1050763, and Byron IR 1049823. Braidwood received a NCV for this issue in their 5/6/2010, NRC Component Design Basis Inspection (CDBI) Report.

In addition to incorporating instrument uncertainty for the CS pumps into the flow analysis, Revision 4 of BYR06-029/BRW-06-0016-M also revised instrument uncertainty values for the CV, SI, and RH pumps to be consistent with the values from calculation PSA-B-98-08. Revision 4 also analyzed the impact of lower RWST internal pressures on the CS pump degradation limits and on the minimum ECCS injection flow rates.

The new ECCS and CS flow results from BYR06-029/BRW-06-0016-M, Revision 4, were then used as input to revise various accident analyses. The common Byron/Braidwood accident analyses were revised in conjunction with a Braidwood change to the maximum UHS supply water temperature (Braidwood EC 396478). The higher new Braidwood input on UHS supply water temperature was conservatively applied to the analyses for Byron.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Units 1 and 2

Activity/Document Number: EC 392420 and DRP 17-034 **Revision Number:** 0/0

Title: ECCS/CS Hydraulic Analyses Incorporate Instrument Uncertainty from CS Pump Testing

For the LOCA mass and energy (M&E) analyses, changes were also made to address Westinghouse Nuclear Safety Advisory Letters (NASLs) NASL-06-6, NASL-11-5, and NASL-14-2. Additional corrections were made to the SATAN78 power shape selection option to select a chopped cosine power shape and to incorporate the WCAP-10325-P-A evaluation methodology for drift flux and break flow with inertia.

The UFSAR change incorporates the results from the revised analyses.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

This activity does not have any impact on the operation of ECCS and CS pumps during normal or accident plant operations. The revised ECCS and CS flow rates have been evaluated for impact on design basis and accident analyses; the results are within the specific acceptance criteria.

The hydraulic analysis determines an acceptable range of performance for the ECCS and CS pumps. To provide a range of performance, for the cases that use maximum pump performance, the performance curves of the ECCS pumps have been enhanced by 3% (10% for the RH pumps) and, for the cases that use minimum pump performance, the pump curves have been degraded by 10%. The performance curves for the CS pumps have also been enhanced by 3%, but the allowed degradation is limited to the value that has been found acceptable in the applicable safety analyses. The maximum allowed degradation for the CS pumps, Train A and B, has been determined in calculation BYR06-029/BRW-06-0016-M, Revision 5A. The acceptable performance ranges will be incorporated as acceptance criteria for the pump performance surveillance testing.

The impact on the PCT due to the revised ECCS flow rates has been evaluated to be an increase of 2 °F for the LBLOCA and 0 °F for the SBLOCA. The PCT evaluations and results were incorporated into the 10CFR50.46 Annual Report.

The revised ECCS flows do not impact the calculated results for a postulated Steam Generator Tube Rupture event. The revised LOCA and Steamline Break inside containment analyses show that the peak calculated containment pressure is below the peak calculated containment internal pressure (P_a), as referenced in Byron Technical Specification 5.5.16. The revised containment temperature profile is bounded by the EQ temperature profile. For postulated steam line breaks outside of containment, the changes had little effect on the limiting transients and the current UFSAR results and figures remain valid.

The changes/corrections made to the LOCA M&E analysis method corrected identified errors and incorporated NRC approved model options. The corrections associated with NASL-06-6, NASL-11-5, and NASL-14-2 result in a penalty to calculated containment pressure and temperature.

The calculated minimum containment flood level remains greater than the minimum required flood level for operation of the Containment Recirculation sumps. The calculated maximum containment flood level remains less than the evaluated containment flood level described in the UFSAR.

The Net Positive Suction Head (NPSH) Available during recirculation operation remains greater than the NPSH Required for the RH and CS pumps. In addition, the void fraction at the pumps remains within the pump acceptance criteria.

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Station/Unit(s): Byron/Units 1 and 2

Activity/Document Number: EC 392420 and DRP 17-034 **Revision Number:** 0/0

Title: ECCS/CS Hydraulic Analyses Incorporate Instrument Uncertainty from CS Pump Testing

The revised ECCS and CS flow rates were determined to have no impact on the RWST level setpoints or the operator action times for manual switchover from injection to recirculation post-LOCA.

ECCS sump downstream wear was re-evaluated. The components evaluated for wear show an acceptable level of wear for a mission time of 30 days. Pump performance remains adequate to meet the required post-LOCA flows.

The revised post-LOCA chemical effects analysis shows that the vendor (CCI) chemical effects testing for the containment sump screens used bounding chemical quantities. The calculation changes do not change the performance of the containment sump screens.

The results of the ECCS/CS hydraulic analysis show that the minimum CS flow to Containment remains above the input for Iodine removal in the LBLOCA dose analysis. With the maximum allowed CS pump degradation (minimum CS pump flows) the CS spray nozzle flow will be lower than the UFSAR described minimum value of 15 gpm per nozzle. The lower spray nozzle flow was evaluated and the resultant drop size remains less than the 1,000 μ limit used in the containment analysis. The small reduction in the spray nozzle flow could result in a reduction in the spray coverage area for each nozzle. The small reduction in spray coverage area for each nozzle will have an insignificant impact on the overall sprayed containment volume used in the dose analysis.

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

Based on the results of the 50.59 Evaluation, the activity proposed in EC 392420 can be implemented without a License Amendment Request.

EC 392420 and the associated UFSAR changes do not introduce the possibility of a change in the frequency of an accident because the calculations that are addressed in EC 392420 evaluate system performance for postulated accidents and are not associated with an initiator of any accident. No new failure modes are introduced due to the results of the calculations. The results of the calculations that use the revised flow rates are acceptable and support post-accident operation of the affected equipment. The UFSAR changes in DRP 17-034 will assure that analyses inputs and assumptions are, and continue to be, complied with.

The activity that is addressed in EC 392420 does not make changes to plant equipment as to change the likelihood of a malfunction of an SSC because the revised calculations evaluate postulated accidents and failures. No new failure modes are introduced. Adequate Net Positive Suction Head (NPSH) Available has been confirmed for the ECCS and CS pumps while taking suction from the Containment recirculation sumps (BRW-06-0035-M/BYR06-058); therefore, this activity does not increase the likelihood of a malfunction of an SSC important to safety. Adequate NPSH from the RWST had already been verified in calculation BYR04-016, Revision 3; this analysis was not affected by the revised ECCS/CS flow rates because the analysis uses runout flow rates. The changes that were made to the GSI-191 Downstream Effects Analysis (BRW-05-063-M/BYR05-061, Revision 4) do not introduce additional debris in the post-LOCA recirculated water or change the sump screens; therefore, the likelihood of a malfunction for ECCS and CS pumps is not increased.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Units 1 and 2

Activity/Document Number: EC 392420 and DRP 17-034 **Revision Number:** 0/0

Title: ECCS/CS Hydraulic Analyses Incorporate Instrument Uncertainty from CS Pump Testing

The calculated minimum containment flood height remains greater than the minimum required flood height for operation of the containment recirculation sumps. The calculated maximum containment flood level remains less than the evaluated containment flood level described in UFSAR Section D3.6.1.2. The revised containment temperature profile is bounded by the Environmental Qualification temperature profile used for qualification of EQ equipment. Thus, the proposed changes do not increase the likelihood of occurrence of a malfunction of equipment important to safety inside containment.

The proposed changes had essentially no impact on the calculated pressure and temperature for postulated steam line breaks outside of containment. Thus, there is a less than minimal increase in the likelihood of malfunction of equipment impacted by a MSLB outside of containment.

The proposed changes do not change the RWST level setpoints, the required time to perform the manual actions to complete switchover from the post-LOCA injection mode to the recirculation mode of operation, or the switchover procedure steps. Thus, the likelihood of a malfunction during switchover is unchanged.

The results of the revised hydraulic analysis show that the Containment Spray flow rates bound the flow that was used in the LOCA dose analysis. The change has an insignificant impact on the input parameters (fraction of containment sprayed and containment volume sprayed) used in the LOCA dose analysis. The revised flow rates for the Steam Generator Tube Rupture event do not have an impact on the results of the dose analysis.

The calculated peak containment pressures are below the current Technical Specification 5.5.16 peak calculated containment pressure value, Pa, and containment design pressure. Thus, containment integrity is unchanged. Fuel integrity is unchanged because the calculated peak clad temperature remains below the design limit.

This activity does not affect the operation of plant equipment. The ECCS and CS pumps will be able to operate as credited in the UFSAR in response to a malfunction of an SSC. The revised CS flow rates have been evaluated and have been found to be higher than the flows used in the dose analyses and the CS spray nozzle drop size remains acceptable. In addition, the peak calculated containment pressure is below the P_a value in Technical Specifications 5.5.16. Therefore, this activity does not increase the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

EC 392420 does not create the possibility for an accident of a different type than any previously analyzed in the UFSAR. The calculations that are addressed in EC 392420 evaluate system performance for postulated accidents and are not associated with an initiator of any accident. No new failure modes are introduced.

The analyses that are implemented with EC 392420 do not introduce the possibility for a malfunction of an SSC with a different result because the activity does not introduce a new failure result that is not bounded by those described in UFSAR Section 6.2.1.3.5 (LOCA), UFSAR Section 6.2.1.4.3 (MSLB), and UFSAR Table 6.5-1 (CS System). The revised analyses use limiting inputs based on the UFSAR described, design basis, postulated failures. These analyses show that the results are acceptable and bounded by existing acceptance criteria. Therefore, a malfunction of an SSC important to safety does not have a different result than any previously evaluated in the UFSAR.

The proposed activity does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. The peak calculated containment pressure is less than the design basis P_a

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Station/Unit(s): Byron/Units 1 and 2

Activity/Document Number: EC 392420 and DRP 17-034 **Revision Number:** 0/0

Title: ECCS/CS Hydraulic Analyses Incorporate Instrument Uncertainty from CS Pump Testing

as given in Byron Technical Specifications 5.5.16. The impact on the Peak Clad Temperature (PCT) due to the revised ECCS flow rates has been calculated to be an increase of 2 °F for the LBLOCA and 0 °F for the SBLOCA. The increase in PCT was included in EC401531, "2015 ANNUAL 10 CFR 50.46 REPORT: BYRON AND BRAIDWOOD UNIT 1 & 2". PCT remains below the UFSAR described design basis limit.

The design analyses changes for EC 392420 do not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses. UFSAR Section D3.6.1 describes the analysis for maximum containment flooding level. The analysis method and assumptions are unchanged. The analysis does not credit water trapped in the refueling cavity consistent with the requirement of Regulatory Guide 1.82. The change to the minimum flood level analysis does not revise the amount of trapped water.

UFSAR Appendix A associated with Regulatory Guide 1.82, discusses that the post-LOCA chemical effects has been evaluated for the sump screens, however the method of evaluation is not specified in the UFSAR or in Regulatory Guide 1.82. The method of evaluation for downstream effects wear is not specified in the UFSAR. The method of evaluation for determining RH and CS pump available NPSH is unchanged. No credit is taken in the NPSH analysis for increase in containment pressure due to the accident. Consistent with Regulatory Guide 1.82 and the UFSAR description, no credit is taken in the NPSH analysis for increase in containment pressure due to the accident. The UFSAR described Computational Fluid Dynamics (CFD) analysis method for calculating containment sump head loss is unchanged. The method of evaluation for calculating RCFC performance is not specified in the UFSAR. UFSAR Section 6.3.5.4 describes the basis for the RWST level setpoints, however the method of evaluation is not specified in the UFSAR.

The revisions for the Steam Generator Tube Rupture (SGTR) Margin to Overfill and the SGTR input to dose analyses revise inputs used in the analyses, and the UFSAR described method of evaluation is unchanged. The outside containment steam line break analyses were revised to evaluate the change in input for ECCS flow and higher UHS supply water temperature when used as the supply for the AF system. The method of evaluation used to determine the main steam tunnel compartment response temperatures for equipment qualification (EQ) was not changed. The method of evaluation for calculating the main steam tunnel EQ temperatures is not specified in the UFSAR.

UFSAR Section 6.2.1.1.3 describes the method for calculating peak containment pressure and temperature. The containment analyses use the COCO software to determine the containment temperature and pressure post-accident. The required CS droplet size in the COCO software is 1000 μ or less. The minimum CS flow rate for each Spray nozzle is 13.7 gpm for Train B. The resulting drop size is less than 1,000 μ. Thus, the change in CS minimum nozzle flow does not result in a departure from the method of evaluation as described in the UFSAR.

The corrections associated with NASL-06-6, NASL-11-5, and NSAL-14-2 were explicitly addressed in the revised analysis. A description of the issues related to the subject NASLs and the potential impact on the LOCA M&E analysis was documented in a RAI response letter for Braidwood (Letter RS-15-129 dated 4/30/2015). The correction of errors to address the issues related to NASLs-06-6, -11-5, and -14-2 were reviewed and accepted as described in the NRC Safety Evaluation for Braidwood License Amendment No. 189. The analysis method and application accepted for Braidwood, is the same analysis method and application being used at Byron. Since the method used has been approved by the NRC for the intended application, the corrections associated with NASL-06-6, NASL-11-5, and NSAL-14-2 are not a departure from a method as described in the UFSAR.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Units 1 and 2

Activity/Document Number: EC 392420 and DRP 17-034 **Revision Number:** 0/0

Title: ECCS/CS Hydraulic Analyses Incorporate Instrument Uncertainty from CS Pump Testing

The SATAN78 input model was revised to include and credit the effects for drift flux and break with inertia consistent with the 2/17/87, NRC SER for WCAP-10325-P, "Westinghouse LOCA M&E Release Model for Containment Design". In addition, the WCAP-10325-P evaluation methodology has been approved to incorporate improved fluid momentum flux terms, which were applied to the break flow model.

The SATAN78 input model was revised to model a chopped cosine power shape so the exit temperature from the upper core node equals the core outlet temperature. This change is per the NRC approved LOCA M&E release methodology in WCAP-10325-P.

The calculation revisions for the inside containment steam line break analyses revised inputs to the analyses and the UFSAR described method of evaluation is unchanged.

All the 50.59 Evaluation questions were answered "No". Thus, the activity may be implemented per plant procedures without obtaining a License Amendment.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☐	**Applicability Review**			
☐	**50.59 Screening**	**50.59 Screening No.**		**Rev.**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-15-001	**Rev.** 0

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron / Unit 1

Activity/Document Number: EC 402344/TS Bases Change 16-008 **Revision Number:** 000

Title: Replace RCP Underfrequency KF Relays with Circuit Shield Type 81 Frequency Relays to Address ABB Part 21 Notification of Potential Defect for KF Relay ZPA

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

The proposed activity replaces the existing Reactor Coolant Pump (RCP) bus underfrequency relays on the four 6.9 kV buses. There are currently 8 existing electromechanical relays (ABB Type KF), two on each bus for an A and B train logic actuation that will be replaced with 8 new solid-state relays (ABB Type 81). These relays provide input to the Reactor Protection System (RPS), where an underfrequency condition on two of the four 6.9 kV buses, determined by either logic train above P-7, produces a reactor trip and a trip of the four reactor coolant pumps.

The current RCP underfrequency relays are powered from the RCP 6.9kV bus Potential Transformers (PT). The new solid state relays will require a DC power input. The power supply required will be installed in this modification.

The Technical Requirements Manual (TRM) currently describes the RCP underfrequency relay nominal setpoint as ≥ 57Hz. The new ABB Type 81 underfrequency relays being installed by this modification will have the same nominal setpoint. The Technical Specification 3.3.1 Allowable Value for the RCP underfrequency relay is ≥ 56.08 Hz; no change to the Technical Specification Allowable Value is required for the new ABB Type 81 underfrequency relays.

The proposed activity will also revise calculations to document numerous loading changes that have been added to the DC system. The calculations determine the voltage drops, required voltages at the safety-related DC components, battery performance for the designed duty cycle and battery charger performance.

- BYR97-204 "125 VDC Battery Sizing Calculation" This calculation is to verify that the ESF 125V DC batteries are adequately sized to supply the design duty cycle in both the normal and cross-tied alignments and ensure a minimum end of duty cycle voltage
- BYR97-205 "125 VDC Battery Charger Sizing Calculation" This calculation is to demonstrate that the existing 400A battery chargers have sufficient capacity to perform their intended functions
- BYR97-224 "125 VDC Voltage Drop Calculation" This calculation is to demonstrate that all safety-related loads from the 125 VDC distribution centers can perform their intended safety functions at the system minimum voltage. This does not include a verification to the AC bus Due ripping and closing coils
- BYR 2000-136 "Voltage Drop Calculation for 4160V Switchgear Breaker Control Circuits" This calculation evaluates the adequacy of the voltage at the 4160V break trip and close coils.
- BYR 2000-191 "Voltage Drop Calculation for 480V Switchgear Breaker Control Circuits" This calculation evaluates the adequacy of the voltage at the 480V break trip and close coils.
- BYR 2000-136 "Voltage Drop Calculation for 6.9kV Switchgear Breaker Control Circuits" This calculation evaluates the adequacy of the voltage at the 6.9kVV break trip and close coils.
- BYR11-003 "Calculation for 56 Cell operation of Batteries 1DC01E, 1DC02E, 2DC01E and 2DC02E" This calculation is to evaluate the capacity margin for operation with less than 58 cells for the unit 1 and unit 2 ESF 125VDC batteries at the end of the duty cycle.

Due to changes in the battery loading the TS bases B 3.8.4 'DC Source-Operating' will be revised to show the new minimum required voltage for the ESF batteries.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

The manufacturer of the existing underfrequency relays has issued a 10 CFR 21 notification (Event # 50691, dated 12/18/14) alerting affected licensees that a review of the seismic qualification of the devices determined that certain styles of the relays do not meet the previously published seismic rating. An engineering review, Operability Evaluation 15-001 (EC 400644), has concluded that the Byron underfrequency relays do not conform to station seismic requirements and should be replaced.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron / Unit 1

Activity/Document Number: EC 402344/TS Bases Change 16-008 **Revision Number:** 000

Title: Replace RCP Underfrequency KF Relays with Circuit Shield Type 81 Frequency Relays to Address ABB Part 21 Notification of Potential Defect for KF Relay ZPA

The battery calculations are being revised to incorporate additional loads to the 125VDC system, and to increase the margin between the calculated minimum 125V DC battery terminal voltage and the required minimum voltage.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

Like the existing relays, the new underfrequency relays will receive the bus frequency input from the existing 6.9 kV bus potential transformers and, on a detected low frequency input into the RPS, will initiate a reactor and RCP trip. The trip setpoint and time delay will remain the same. The new relays will be powered from 125 VDC ESF distribution panels, whereas the existing relays are powered only from the 6.9 kV bus potential transformers. The change in power supply to the relays introduces a new failure mechanism and is considered as adverse in the Screening.

The revised calculations listed above demonstrate that after the addition of the new DC loads the battery terminal voltage will be acceptable to support the operation of the safety-related loads. In addition the revised BYR11-003 "Calculation for 56 Cell operation of Batteries 1DC01E, 1DC02E, 2DC01E and 2DC02E" demonstrates that with less than 58 cells the batteries will be able to meet the design duty cycle requirements .There is no impact on the design bases, UFSAR described methodology or safety analyses described in the UFSAR by the listed calculation revisions.

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

The proposed change replaces the existing RCP electro-mechanical underfrequency relays with new solid state underfrequency relays. The change is required because the existing relays do not meet the seismic requirements. This change will require a new 125V DC power supply to be provided for the new relays. The existing nominal setpoint and T.S allowable value will be retained.

The RCP underfrequency relays are required to provide input for a reactor trip and a trip of all four RCPs on a detected low frequency. The new relays do not change the method of performing these functions. The result of a detected low frequency remains unchanged.

The existing relays use the bus PTs for both frequency input and power. The new relays will continue to use the bus PTs for the frequency input; however, they will require a new 125 V DC power supply. This introduces a new failure mechanism: loss of DC power to an underfrequency relay. If the DC power is lost the relay will fail to the 'not tripped' condition. This provides a reduction in reliability of the relay to perform its function, and this is considered to be an adverse impact. The new failure mechanism was reviewed in the 50.59 evaluation and determined to result in no more than a minimal increase in the likelihood of a malfunction of an SSC important to safety previously evaluated in the UFSAR. This was due to the reliability of the DC power system, the fact that failure of the DC power supply is alarmed in the control room and the ability of the operators to establish the redundant DC power supply.

The methodology used in the calculation for determining the 125VDC Battery sizing to meet the design duty cycle follows the methodology described in the UFSAR for this determination. The methodology used in the calculation to determine the battery charger size follows the methodology described in the UFSAR.

Therefore, based on this Screening and Evaluation the activity may be implemented under the governing procedures.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron / Unit 1

Activity/Document Number: EC 402344/TS Bases Change 16-008 **Revision Number:** 000

Title: Replace RCP Underfrequency KF Relays with Circuit Shield Type 81 Frequency Relays to Address ABB Part 21 Notification of Potential Defect for KF Relay ZPA

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	**6E-16-073**	**Rev.**	**0**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-16-004**	**Rev.**	**0**

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 1/Unit 2

Activity/Document Number: 0BOA PRI-7 **Revision Number:** 2

Title: 0BOA PRI-7 Loss of Ultimate Heat Sink Unit 0

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

The activity is to revise 0BOA PRI-7, "Loss of Ultimate Heat Sink Unit 0", to allow the use of the non-safety related piping connection (1/2 FX307B-6) between the unit 1/2 'B' Essential Service Water (SX) pumps and the 'B' Auxiliary Feed Water (AFW) Shaft Driven Cooling Water Pumps (1/2 SX04P) in the beyond-design-basis event where there is a complete loss of ultimate heat sink cooling capability. This flow path was installed under EC 393374 to support the requirements of the Flexible and Diverse Coping Strategies (FLEX) Implementation Guide, NEI 12-06 and NRC Order EA-12-049, Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design Basis External Events.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

Currently Guideline 1/2 BFSG-2 "Alternate AFW/EFW Suction Source" for Unit 1 and Unit 2 is approved to allow the use of the FLEX piping FX307B-6 to connect the discharge of the 'B' SX pumps to the suction of the 'B' AFW pump following a loss of all AC power if the CST is not available and an alternate suction source for the 'B' AFW pump is required. In order to enable the probabilistic risk assessment (PRA) evaluations for Byron Station to credit this connection during a beyond-design-basis event of a loss of all SX, a station procedure is required to implement the actions to connect the SX alternate flow path to the AF Shaft Driven Cooling Water Pump. The Operating Abnormal Procedure 0BOA PRI-7 currently exists to address a complete loss of ultimate heat sink cooling capability; however, it does not currently credit the use of the FLEX alternate flow path connection piping FX307B.

Currently, upon a loss of all SX pumps the 'B' AFW Shaft Driven Cooling Water pump takes suction from the SX system, pumps it through the 'B' AFW pump room cooler, engine cooler, and gear coolers to maintain operation of the 'B' AFW pump. However the discharge of the Shaft Driven Cooling Water pump cycles back into the suction of the pump due to the hydraulic pressures in the SX system. This creates a 'cycling' effect where the cooling water being used by the 'B' AFW pump is being heated by the waste heat from the pump. The current design documents demonstrate the 'B' AFW pump can operate for a minimum of 2 hours in this condition. For extended operation of the 'B' AFW pump following this beyond-design basis event (a complete loss of ultimate heat sink cooling capability) a new cooling source is needed. This activity, revising 0BOA PRI-7, provides guidance on aligning the FLEX alternate path to provide a new cooling source. When the FLEX alternate path is aligned, the Shaft Driven Cooling pump will take suction for the Essential Service Cooling Water Basin and discharge back to the basin, thus ensuring long term access to sufficient cooling water to support the extended operation of the 'B' AFW pump.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

The effect of this activity is to allow alignment the non-safety related FLEX alternate flow path between the SX pump discharge and the 'B' AFW Shaft Driven Cooling water pump suction following a complete loss of ultimate heat sink cooling capability. This allows for extended, greater than 2 hour, run times for the 'B' AFW pump in the condition where there is no forced SX flow on either Unit. The AFW system has a safety related design function to operate during a loss of all AC – a condition which also results in a loss of forced SX flow. The time period for 'B' AFW pump operation during a loss of all AC is described in the UFSAR as being a minimum of 2 hours.

Connecting the non-safety related FLEX piping to the safety-related AFW system following a complete loss of ultimate heat sink cooling during this 2 hour period adversely affects the AFW system. The non-safety related FLEX piping has been designed (as described in EC 393374) as Augmented Quality, seismic category I and is designed to remain functional after a Design Basis Earthquake. There the piping is considered no likely to fail during the maximum 2 hour operation it will be required to support the AFW pumps design requirement. In addition a failure of the non-safety related piping such that it can no longer support operation of the 'B' AFW pump is prevented from damaging the pump due to the existing high temperature trips on the pump.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): **Byron Unit 1/Unit 2**

Activity/Document Number: 0BOA PRI-7 **Revision Number: 2**

Title: 0BOA PRI-7 Loss of Ultimate Heat Sink Unit 0

Therefore, the safety related piping connection remains available if needed to replace a degraded non-safety related FLEX piping. The Operator will make the determination that the condition where there is no SX forced flow will exist for longer than an hour, and if it will then the FLEX piping will be used, thus ensuring the safety related piping remains available to support pump operations for as long as practical.

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

The design requirement of the AFW system, for the loss of all AC event, is to provide a safety related injection to the steam generators for a period of at least 2 hours. This activity permits the use of a non-safety related piping connection to the AFW system following a beyond-design-basis event (complete loss of ultimate heat sink cooling capability) to allow extending the operation time of the AFW system beyond the 2 hour requirement, however this piping system may be connected prior to the 2 hour operation time being exceeded.

The piping has been designed to withstand a design basis seismic event and continue to function. The water source for the Shaft Driven Cooling pump will be changed from the outlet of the SX strainer, to the inlet of the strainer, and the effects of this non-strained water has been evaluated to not degrade the AFW pump performance. The use of procedure 0BOA PRI-7 to direct the evolution to connect the FLEX alternate piping has been reviewed and determined to be within the allowed use for an Operating Abnormal Procedure as described in UFSAR section 13.5 'Plant Procedures'. However, the AF system currently is designed to start, initiate and provide injection flow to the Steam Generators without operator intervention. The change to 0BOA PRI-7 requires operator intervention, albeit an optional intervention, that has been determined to be adverse in the screening.

In addition allowing the connection of the non-safety related FLEX piping to the AFW system during the time period where the AFW system is described as performing a safety related function is determined to be adverse in the screening, and will be reviewed in the 50.59 Evaluation.

The attached Evaluation has demonstrated that the effect of using the non-safety FLEX piping does not present a more then minimal increase in the likelihood of a malfunction of an SSC important to safety previously evaluated in the UFSAR. The use of the non-safety related seismic Flex piping, as been determined to be adverse in the attached screening. This adverse condition was determined to be a not more than minimal increase in the likelihood of a malfunction of an SSC. In addition the operator intervention provided by the procedure has been demonstrated to not be more than a minimal increase in the likelihood of a malfunction of an SSC due to the option to use operator action to extend the operational time for the AFW system during a loss of all forced SX flow.

Therefore, based on this Screening and Evaluation the activity may be implemented under the governing procedures.

This activity is determined to be adverse, based on connecting a non-safety related piping section to the AFW during the time period (2 hour runtime requirement) where the AFW system is performing a safety related activity during a loss of all AC.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	**6D-16-009**	**Rev.**	**0**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-16-005**	**Rev.**	**0**

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): **Byron Unit 1/Unit 2**

Activity/Document Number: 0BOA PRI-7 **Revision Number:** 2

Title: **0BOA PRI-7 Loss of Ultimate Heat Sink Unit 0**

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 406958 **Revision Number:** 0

Title: Temporarily Defeat FW Water Hammer Prevention System (WHPS) FW Isolation Signals During Normal Power Operation For Steam Generators 2A/2B/2C/2D

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

This Temporary Configuration Change (TCC) Engineering Change (EC) 406958 will disable the Water Hammer Prevention System (WHPS) Steam Generator low level and Steam Generator low pressure feedwater isolation signals on Unit 2 after Feedwater (FW) flow is transferred from the main feedwater isolation valve bypass valves, 2FW043A/B/C/D, to the main feedwater isolation valves, 2FW009A/B/C/D, (when power level is above approximately 25%).

The proposed change to disable the WHPS FW isolation involves lifting a lead from a terminal block for each of the 2FW009A/B/C/D valves, installing an electric switchable jumper between two terminal block points for each of the 2FW039A/B/C/D valves, and repositioning the control switch from "Auto" to "Open" for each of the 2FW035A/B/C/D valves. The WHPS FW isolation signals to the 2FW043A/B/C/D valves are not disabled because these valves are normally in the closed position above 25% power level.

The following changes will be made to plant operating procedures:

- 2BGP 100-1, PLANT HEATUP, will be revised to verify that WHPS FW isolation has been enabled prior to starting any FW pumps.
- 2BGP 100-3, POWER ASCENSION, will be revised to disable WHPS FW isolation after the 2FW009A/B/C/D valves are open.
- 2BGP 100-4, POWER DESCENSION, will be revised to enable WHPS FW isolation for the 2FW035A/B/C/D and 2FW039A/B/C/D valves prior to closing the 2FW009A/B/C/D valves, and then perform a follow-up action to enable WHPS FW isolation on the 2FW009A/B/C/D valves.
- 2BGP 100-4T4, REACTOR TRIP POST RESPONSE GUIDELINE, will be revised to verify that the 2FW009A/B/C/D, 2FW043A/B/C/D, and 2FW039A/B/C/D valves are closed prior to starting the Startup Feedwater Pump. Action will also be taken to enable WHPS FW isolation later in the post trip response.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

This activity provides the capability to disable the WHPS Steam Generator low level and Steam Generator low pressure feedwater isolation signals during power operation above start-up levels to address an adverse vulnerability with the existing design. The existing design can cause spurious/unnecessary WHPS actuation, feedwater isolation, loss of normal feedwater flow, and reactor trip solely due to the loss of a single non-safety related power supply for the relays.

The changes to the plant operating procedures are made to administratively control when the WHPS FW isolation function is disabled and enabled.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

The safety function of the Feedwater Isolation Valves is to close, isolating main feedwater flow upon a safety injection signal, feedwater isolation signal and high-2 steam generator level. As described in the Byron UFSAR, "Several water hammer prevention features have been designed into the feedwater system. These features are provided to minimize the possibility of various water hammer phenomena in the steam generator preheater, steam generator main feedwater inlet piping, and the steam generator upper nozzle feedwater piping." Steam generator (two-out-of-three logic) low level trips are provided to close all feedwater isolation valves, feedwater isolation bypass valves and feedwater preheater bypass valves. Steam generator (two-out-

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 406958 **Revision Number:** 0

Title: Temporarily Defeat FW Water Hammer Prevention System (WHPS) FW Isolation Signals During Normal Power Operation For Steam Generators 2A/2B/2C/2D

of-three logic) low pressure trips are provided to close all feedwater isolation valves, feedwater isolation bypass valves, feedwater preheater bypass valves, and the feedwater bypass tempering valves."

The UFSAR described water hammer prevention features were included in the NRC's basis for approval of (1) the designs of the Steam Generator and the FW/AFW systems, and the measures to address Unresolved Safety Issue (USI) A-1, Steam Generator Water Hammer; and (2) the removal of pipe restraints resulting from the elimination of arbitrary intermediate breaks in the FW system piping system design.

The purpose of the WHPS FW isolation is to prevent water hammer by precluding the admission of cool FW into a steam-voided preheater section of the steam generator. When the WHPS FW isolation is disabled at power, the safety related ESFAS signals function to close the feedwater isolation valves on low steam line pressure safety injection or steam generator water level low-low reactor trip coincident with Tave less than 564° F. A review of the postulated transients that could create conditions for a water hammer event when power is above 25% was performed (Refer to EC 406958). The review determined that the ESFAS FW isolation signals (Steam line pressure SI or the Reactor trip + Lo Tave) will function to prevent water hammer by precluding the admission of cool FW into a steam-voided preheater section of the steam generator during power operating conditions. Note that the 2FW039 valves close on reactor trip and no coincident Tave is required. At low power levels or during startup the WHPS FW isolation features will be enabled. The design bases function to preclude the admission of cool FW into a steam-voided preheater section of the steam generator is not adversely impacted.

The other UFSAR described WHPS features will not be altered by this EC. Operating procedures associated with only providing feedwater to the upper nozzle of the steam generator during startup and low load conditions are unchanged. Temperature monitoring and alarms to detect possible back leakage of steam from the steam generators into the feedwater piping are unchanged. Forward flushing and temperature permissive interlocks associated with opening the FWIV are also unchanged. This EC does not alter split feedwater flow operation.

Implementation of EC 406958 does not impact the safety related circuits described in the UFSAR and Technical Specifications that generate an independent Feedwater Isolation Signal, including Safety Injection (SI), SG Level HI-2 Trip (P-14), and Reactor Trip (P-4) coincident with RCS Average Temperature Low at 564°F. Implementation of this EC does not impact the Reactor Trip signal generated on low-low steam generator water level.

The change will require new operator actions to appropriately enable/disable WHPS FW isolation in the plant startup, power ascension, power descension, and plant trip response procedures. Operators will reposition the control board hand switch for the 2FW035A/B/C/D valves and reposition the jumper switch for 2FW039A/B/C/D valves. Operations will also request Maintenance to either lift or re-land the leads for the 2FW009A/B/C/D valves.

- For startup WHPS FW isolation will be verified to be enabled prior to starting any FW pumps. Once the 2FW009 valves are open, action will be taken to disable WHPS FW isolation for the 2FW009, 2FW035, and 2FW039 valves.
- For a normal shutdown or low power operation action will be taken to re-enable WHPS FW isolation for the 2FW035 and 2FW039 valves prior to closing the 2FW009 valves. WHPS FW isolation for the 2FW009 valves will be enabled prior to defeating the ESFAS FW isolation signals.
- On a plant trip action will be taken to verify that the 2FW009, 2FW039, and 2FW043 valves are closed prior to starting the Startup FW pump. The 2FW035 valve flow path is used to direct FW to the upper nozzle. The UFSAR described main FW flow and temperature permissive interlocks will prevent inadvertent introduction of cold FW to the lower nozzle until action is taken to enable WHPS FW isolation.

There is no impact on any emergency operating procedures. The proposed operator actions are not time critical and do not impose a significant new burden on the operating crew.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 406958 **Revision Number:** 0

Title: Temporarily Defeat FW Water Hammer Prevention System (WHPS) FW Isolation Signals During Normal Power Operation For Steam Generators 2A/2B/2C/2D

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

Above 25% power the safety related ESFAS signals function to close the feedwater isolation valves on a low steam line pressure safety injection (< 640 psig) or steam generator water level low-low reactor trip (≤ 36.3% NR) coincident with low Tave (less than 564° F). Note that the 2FW039A/B/C/D valves close on reactor trip and no coincident Tave is required. A review of the postulated transients that could create conditions for a water hammer event when power is above 25% was performed (Refer to EC 406958). The review determined that the ESFAS FW isolation signals (Steam line pressure SI or the Reactor trip + Lo Tave) will function to prevent water hammer by precluding the admission of cool FW into a steam-voided preheater section of the steam generator during power operating conditions. Therefore disabling the WHPS low level and low pressure FW isolation auto close signals above 25% power does not increase the probability of water hammer in the SG preheater and the main FW inlet piping. Thus the proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident caused by a water hammer event in the FW line or the Steam Generator (FW line break or steam generator tube rupture).

The defeated WHPS FW isolation function is not used or credited to mitigate any design basis event, analysis, or Emergency Operating Procedure Operator Action. The credited automatic function is the ESFAS FW Isolation which is a completely separate circuit. Thus the ESFAS FW Isolation is not affected by this change.

The change adds new operator actions to disable and enable the WHPS low level and low pressure FW isolation auto close signals. Failure to enable the WHPS low level and low pressure auto close signals for startup, low load, or increasing load conditions could potentially result in a malfunction of equipment important to safety, specifically prevention of water hammer in the steam generator preheater and/or steam generator main feedwater (FW) inlet piping. The likelihood of occurrence of a malfunction associated with the new operator actions is not more than minimal based on the following:

- The actions to disable/enable the WHPS low level and low pressure FW isolation auto close signals has been added to the appropriate plant procedures and operators will be trained on the revised procedures.

- The actions to disable/enable the WHPS low level and low pressure FW isolation auto close signals do not have a required time for completion. Above 25% power there is no specific time requirement for disabling the WHPS low level and low pressure FW isolation auto close signals. There is no impact on any emergency operating procedures. The proposed operator actions for restart, power descension, and reactor trip are not time critical and do not impose a significant new burden on the operating crew.

- This change does not introduce credible errors during performance of the operator actions required for this change. Three actions are required to block or enable WHPS following this design change. Lifting a single lead for each 2FW009 valve in 2PA27 or 2PA28J panel. These leads will be labeled as part of this design change. The terminal board is clearly and accurately labeled. Manipulation of wires on a terminal board is considered a routine action with no credible failure risk. Lifting/landing the wire does not initiate a water hammer prevention actuation; it enables/blocks the circuit. The 2FW039A/B/C/D valve water hammer prevention feature will be enabled or blocked with a switchable jumper. This jumper will be installed across the original water hammer prevention actuation relay contacts, thereby allowing blocking or enabling the original contacts. This switch will be clearly labeled. The 2FW035A/B/C/D valve control switch in the Main Control Room bypasses the water hammer prevention circuit when it is placed in open (as opposed to auto). This is a routine method of configuration control by operators.

  Lifting/landing of these leads for 2FW009's, operation of the blocking switch for the 2FW039's, and operation of the permanent switch for the 2FW035's will be controlled as part of a formal procedure. The wires to be lifted, switchable jumpers to be installed and used, and hand switches to be manipulated per this activity will be clearly flagged and labelled per the temporary Engineering Change to allow for clear and easy operator identification. Maintenance procedure MA-AA-1070, which maintenance has trained on, exists to provide guidance for lifting and

Station/Unit(s): **Byron Unit 2**

Activity/Document Number: **EC 406958** **Revision Number:** **0**

Title: **Temporarily Defeat FW Water Hammer Prevention System (WHPS) FW Isolation Signals During Normal Power Operation For Steam Generators 2A/2B/2C/2D**

landing leads and installing jumpers and will be utilized in support of this activity. These actions do not initiate a water hammer prevention actuation; therefore, there would be time to recover from unlikely actions such as blocking or enabling WHPs out of the expected sequence.

- The system function to preclude the admission of cool FW into a steam-voided preheater section of the steam generator is not adversely impacted.

Above 25% power the ESFAS FW isolation signals will function to prevent water hammer by precluding the admission of cool FW into a steam-voided preheater section of the steam generator. Appropriate administrative controls will be put in place to enable the automatic WHPS FW isolation function during plant startup, power descension, and after a plant trip when the ESFAS FW isolation is not available. Thus the proposed change does not permanently substitute manual action for automatic action for performing a UFSAR-described design function.

The WHPS low level and low pressure FW isolation auto close signals are classified as non-safety related. The WHPS FW isolation auto close signal for each individual generator is based on a two-out-of-three logic. The safety related ESFAS FW isolation signal on low steam line pressure is based on two-out-of-three logic in any one of the four steam lines. A reactor trip is actuated on two-out-of-four low-low water level signals occurring in any steam generator. A FW isolation signal is generated with the reactor trip coincident with a two-out-of-four Lo Tave signals. The safety related ESFAS FW isolation signals provide the same level of redundancy as the WHPS FW isolation signals. The safety related ESFAS components are considered to be more reliable than the non-safety related WHPS components. The highest risk for water hammer events is during startup and low power operation when the ESFAS FW isolation signals may be bypassed and only the WHPS FW isolation signals would be available for water hammer protection. This same level of redundancy will be in place with only the ESFAS FW isolation signals providing protection when the WHPS low level and low pressure auto close FW isolation signals are disabled above 25% power.

The proposed change involves lifting a lead from a terminal block for each of the 2FW009A/B/C/D valves, installing an electrical switchable jumper between two terminal block points for each of the 2FW039A/B/C/D valves, and repositioning the control switch from "Auto" to "Open" for each of the 2FW035A/B/C/D valves.

In accordance with procedure MA-AA-1070, immediately after the lead is lifted for the 2FW009 valves electrical insulated tape will be applied on the exposed part of the connection. This will prevent impact to the associated 2FW009 valves or any of the other equipment within the panels. Thus the proposed change will not result in a more than minimal increase in the likelihood of occurrence of a malfunction of the 2FW009 valves.

The switchable jumpers used to defeat water hammer protection on the 2FW39A/B/C/D valves are built using a rocker switch and Pomona/banana jacks. These devices are built to handle much more current than the 3 Amp protection fuse for this circuit. The switchable jumpers are put in parallel to the WHPS contacts. In the event of a failure of a cable or connection, it would be a conservative failure that would leave WHPS in an enabled state. The Pomona cables have spring-loaded covers which automatically cover the exposed male connection when the cable is removed from the jack. This will prevent the cable from shorting and closing the 2FW39A/B/C/D valves if the Pomona cable is inadvertently removed from the Pomona jack. The switches used are similar to switches used throughout the plant and have similar reliability. Thus the proposed change does not change the likelihood of occurrence of a malfunction of the 2FW039 valves.

The control switches for the 2FW035 vales are designed to allow the valves to be held open. Thus the proposed change does not change the likelihood of occurrence of a malfunction of the 2FW035 valves.

The proposed change does not impact the other UFSAR described WHPS features. When the WHPS FW isolation is disabled at power, the water hammer prevention function to preclude injection of cool water into a steam-voided preheater section of the steam generator will continue to be performed by the safety related ESFAS signals to close the feedwater isolation valves on low steam line pressure safety injection or steam generator water level low-low reactor trip coincident with Tave less than 564° F. At low power levels or during startup the WHPS FW isolation features will be enabled. Additionally, the proposed change does not revise the operating procedure steps associated with switching feedwater flow from the top feed nozzle to the

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 406958 **Revision Number:** 0

Title: Temporarily Defeat FW Water Hammer Prevention System (WHPS) FW Isolation Signals During Normal Power Operation For Steam Generators 2A/2B/2C/2D

lower main feed nozzle. As described in the response to FSAR Question 010.51 initial plant startup testing was performed to confirm that no damaging water hammer occurs when FW delivery is transferred from the top SG nozzle to the main feed nozzle. Since the transfer procedure steps are unchanged, there is no change in the frequency of a water hammer when FW delivery is transferred. Therefore the proposed change will not result in more than a minimal increase in the likelihood of a water hammer induced malfunction. The results of the Start-up testing are not impacted.

Disabling the WHPS low level and low pressure FW isolation auto close signals above 25% power does not introduce the possibility of a change in the consequences of an accident because the FW isolation from the WHPS is not credited with mitigating any accident previously evaluated in the UFSAR. The ESFAS FW isolation signals (Steam line pressure SI or the Reactor trip + Lo Tave) will continue to close the feedwater isolation valves as designed for accident mitigation. Thus the proposed activity will not result in a more than minimal increase in the consequences of an accident previously evaluated in the UFSAR.

Disabling the WHPS low level and low pressure FW isolation auto close signals above 25% power does not introduce the possibility of a change in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR because the FW isolation from the WHPS is not credited to mitigate the consequences of malfunctions of an SSC important to safety. The proposed change does not impact the safety related ESFAS FW isolation circuitry. Thus there is no change to the UFSAR described failure modes and effects for the ESF actuation system. The proposed changes to the WHPS FW isolation auto close signals does not introduce any new failure mode for the system.

Disabling the WHPS low level and low pressure FW isolation auto close signals above 25% power does not introduce the possibility of a new accident because the proposed change is not the initiator of any accident and no new failure modes are introduced. When the WHPS FW isolation is disabled at power the safety related ESFAS signals will result in FW isolation actions that provide the water hammer prevention function equivalent to the WHPS FW isolation. Administrative controls will be put in place to ensure that the WHPS FW isolation signals are enabled during startup and low power operation. The likelihood of operator error leading to an accident of a different type is not credible because the actions are reflected in plant procedures and operator training programs, the actions can be completed in the required time, the actions are simple and adequate time is available to recover from any credible error. The design bases function to preclude the admission of cool FW into a steam-voided preheater section of the steam generator is not adversely impacted, thus no new accident types are introduced.

Disabling the WHPS low level and low pressure FW isolation auto close signals above 25% power does not introduce the possibility of a malfunction with a different result because when the WHPS FW isolation is disabled at power the water hammer prevention function will be performed by the safety related ESFAS signals. Administrative controls will be put in place to ensure that the WHPs FW isolation signals are enabled during startup and low power operation.

This temporary change to the Unit 2 WHPS Steam Generator low level and Steam Generator low pressure FW isolation auto close signals does not result in a change that would cause any system parameter to change. WHPS is not credited for mitigation in any accident analysis, thus there is no impact on any UFSAR described fission product barrier limit. Therefore, the proposed activity does not result in a Design Basis Limit for a Fission Product Barrier (DBLFPB) as described in the UFSAR being exceeded or altered.

The proposed change does not involve a method of evaluation as defined in LS-AA-104-1000. Therefore, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analysis.

Based on the results of the 50.59 Evaluation the activity may be implemented per plant procedures without obtaining a License Amendment.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): **Byron Unit 2**

Activity/Document Number: **EC 406958** **Revision Number:** **0**

Title: **Temporarily Defeat FW Water Hammer Prevention System (WHPS) FW Isolation Signals During Normal Power Operation For Steam Generators 2A/2B/2C/2D**

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☐	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-16-007	**Rev.**	0

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

Evaluation No. 6G-16-008

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Station/Units 1 and 2

Activity/Document Number: DRP 16-083/TSB Change 16-011 **Revision Number:** 0

Title: Appendix J Scope Reduction as a Result of True North Engineering Report Commissioned by Corporate Engineering

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

The proposed activity is a documentation change addressing the exclusion from 10 CFR 50 Appendix J, Option B, Type C local leak rate testing (LLRT) of the following Containment Isolation Valves:

Penetration	Valve EPN	Description
Chemical and Volume Control System (CV)		
P-41	1(2)CV8152	AOV U-1(U-2) LTDWN HDR ISOL VLV (EOP VLV)
	1(2)CV8160	AOV U-1(U-2) LTDWN ORIFICES OUTLET HDR ISOL VLV (EOP VLV)
P-28	1(2)CV8100	MOV U-1(U-2) RC PPS SEAL L/O HDR OUTSIDE CNMT ISOL VLV
	1(2)CV8112	MOV U-1(U-2) RC PPS SEAL L/O HDR INSIDE CNMT ISOL VLV
	1(2)CV8113	U-1(U-2) RCP SEAL L/O HDR ISOL VLV 1(2)CV8112 BYP HDR CHK VLV
Containment Spray System (CS)		
P-1	1(2)CS007A	MOV 1(2)A CS PP DSCH HDR ISOL VLV
	1(2)CS008A	1(2)A CS PP 1(2)CS01PA SPRAY NOZL HDR CHK VLV
P-16	1(2)CS007B	MOV 1(2)B CS PP DSCH HDR ISOL VLV
	1(2)CS008B	1(2)B CS PP 1(2)CS01PB SPRAY NOZL HDR CHK VLV

UFSAR Section 6.2.4.2.7 will be revised to identify EC 404972 as the source document justifying the exclusion from 10 CFR 50 Appendix J, Option B, Type C local leak rate testing for the penetrations and valves above. UFSAR Table 6.2-58 will be revised to indicate that Type C testing is not applicable for these penetration and valves.

Technical Specifications Bases Table B3.6.3-1 will be revised to annotate that the penetrations and valves above are excluded from 10 CFR 50 Appendix J, Option B, Type C local leak rate testing.

The following procedures are impacted by the exclusion from 10 CFR 50 Appendix J, Option B, Type C local leak rate testing:

1) BAP 1600-11A1
2) BVP 800-39
3) 1/2BVSR 6.1.1-24
4) 1/2BVSR 6.1.1-26
5) 1/2BOSR 6.1.1-9
6) 1/2BOSR 6.1.1-14

Reason for Activity:
(Discuss why the proposed activity is being performed.)

The elimination of local leak rate testing for valves which meet the exclusion criteria will reduce personnel radiation exposure and will decrease work scope for refueling outages.

Evaluation No. 6G-16-008

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): **Byron Station/Units 1 and 2**

Activity/Document Number: **DRP 16-083/TSB Change 16-011** **Revision Number:** **0**

Title: **Appendix J Scope Reduction as a Result of True North Engineering Report Commissioned by Corporate Engineering**

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

This activity revises the UFSAR, Technical Specification Bases, and applicable procedures to reflect the exclusion from 10CFR50, Appendix J, Option B, Type C local leak rate testing for the containment isolation valves listed in the Description of Activity section above. EC 404972 and supporting documents justify this exclusion based on the current licensing basis for Byron Station. The basis for this exclusion is given below.

Technical Specification 5.5.16, Containment Leak Rate Testing Program states that the program shall be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995 with exceptions as noted in the TS. RG 1.163 was issued by the NRC to endorse the use of NEI 94-01.

In NEI 94-01 Revision 0, Section 6.0, General Requirements, components that may be excluded from the LLRT requirements of 10CFR50 Appendix J are described below:

An LLRT is a test performed on Type B and Type C components. An LLRT is not required for the following cases:

- Primary containment boundaries that do not constitute potential primary containment atmospheric pathways during and following a Design Basis Accident (DBA);
- Boundaries sealed with a qualified seal system; or,
- Test connection vents and drains between primary containment isolation valves which are one inch or less in size, administratively secured closed and consist of a double barrier.

EC 404972 demonstrates that the CV and CS valves in the Description of Activity section above satisfy the criteria for exclusion from Type C tests. The valves will continue to be tested in accordance with Inservice Testing (IST) Program required by 10 CFR 50.55a. The affected valves will also remain classified as containment isolation valves. No physical work is required to the valves or the system.

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

A 50.59 Screening was performed which concluded that the activity requires a 50.59 Evaluation to implement the changes. The changes to the documents implementing the LLRT scope reduction under EC 404972 could be considered a change to a procedure (LLRT program implementation procedures) that may adversely affect how a UFSAR described design function is performed or controlled. Because the scope of Type C LLRT testing is being reduced, the effect of the proposed changes is conservatively considered to be adverse, for the purposes of evaluation.

A 50.59 Evaluation was performed which concluded that that the documentation changes can be implemented per plant procedures without obtaining a License Amendment. The proposed changes to the UFSAR, Technical Specification Bases, and applicable procedures do not require a change to Technical Specifications because they do not alter or modify any existing Technical Specification. The basis for this conclusion is presented below:

The proposed activity of revising the UFSAR, Technical Specification Bases, and applicable procedures do not physically alter any equipment, system performance, or operator actions that could affect the accidents and transients in Chapters 6 and 15 of the UFSAR or their frequency of occurrence. Therefore, the documentation changes related to implementing the LLRT scope reduction under EC 404972 do not introduce the possibility of a change in the frequency of occurrence of an accident previously evaluated in the UFSAR because the leakage of these valves is not an in initiator of any accident and no new failure modes are introduced.

Evaluation No. 6G-16-008

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Station/Units 1 and 2

Activity/Document Number: DRP 16-083/TSB Change 16-011 **Revision Number:** 0

Title: Appendix J Scope Reduction as a Result of True North Engineering Report Commissioned by Corporate Engineering

The proposed documentation changes do not physically alter any equipment, system performance, or operator actions that could affect the performance of the design function of the containment isolation valves associated with these changes. The current UFSAR analyses remain bounding. Therefore, the documentation changes implemented under this activity do not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The proposed activity does not result in any changes to the operation of the affected systems (CV and CS) that could impact the diversity and redundancy of those systems in a manner that could reduce the ability to perform their intended design functions. Therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

The affected CV and CS penetrations and valves are excluded from Type C LLRT on the basis that they do not constitute a potential primary containment atmospheric pathway during a design basis accident, therefore, this activity does not result in an increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The documentation changes implemented under this activity do not physically alter any equipment, system performance, or operator actions that could affect the performance of the design function of the containment isolation valves associated with these changes. The current UFSAR analyses remain bounding. Therefore, the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

Removing Type C (LLRT) testing requirements for the affected CV and CS penetrations and containment isolation valves in accordance with approved regulatory documentation does not introduce the possibility for a malfunction of an SSC with a different result because the activity does not introduce any new failure modes or failure results. Therefore, the proposed activity does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

Type C LLRT is being removed on the basis that the valve cannot provide a leakage pathway for containment atmosphere, and therefore ensures that La, the maximum allowable leakage rate, is not impacted. This activity does not result in a change that would cause any system parameter to change. Therefore, the activity does not result in a fission product barrier (and/or a DBLFPB) as described in the UFSAR being exceeded or altered.

The changes to the UFSAR, Technical Specification Bases, and procedures related to implementing the LLRT scope reduction under EC 404972 for the affected CV and CS penetrations and valves do not involve a method of evaluation as described in the UFSAR. La is not being changed as a result of this activity.

Therefore, the activity does not result in a departure from a method of evaluation. This activity does not impact, revise or replace the methods used to determine primary containment leakage.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	6E-16-097	**Rev.**	0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-16-008	**Rev.**	0

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron / Unit 0,1

Activity/Document Number: EC 618443 **Revision Number:** 0

Title: Lost SFP Crimps and RVLIS Pins

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

During the performance of an annual spent fuel pool (SFP) lighting inspection, Reactor Services discovered that five lighting lanyard safety clip crimps are missing (IR 3975511). The crimps are made of 304 stainless steel with dimensions of approximately 0.375"x0.137"x0.229" (Qty. 5) oval crimp fittings. Reactor Services performed an underwater search, but was unsuccessful in locating the missing parts. The activity is to evaluate the effects of this potential foreign material in the Reactor Coolant System (RCS) and Spent Fuel Pool (SFP). Additionally, two locking pins were found missing from RVLIS connectors. These are 0.170" long x 0.1" diameter shaft with a 0.125" diameter head and are made of stainless steel. These were lost in the refuel cavity area.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

Loose parts (5 stainless steel lanyard crimps) are missing and potentially dropped into the Spent Fuel Pool (SFP). Based on IR 3975511, these lost crimps were located in the northeast corner of the spent fuel pool over discharged fuel. These could not be located and are potential foreign material that could have fallen to the bottom of the SFP, or could be on or in a fuel assembly. A likelihood exists for some or all of the crimps to be transferred to the Unit 1 core during B1R21 fuel reload. Additionally, two very small RVLIS locking pins were lost in the refuel cavity area (IRs 3979800 and 3981072). These were located at the drawbridge from the refuel cavity to the reactor head. A lost parts evaluation is required per ER-AA-2006 to evaluate the potential consequences that the missing crimps and pins may have on plant equipment.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

There is no effect on the operation of any system. The location of the five small crimp fittings is not known other than they came from light lanyards along the northeast corner of the spent fuel pool over discharged fuel. There is a small potential that these crimps could be transported into the Unit 1 RCS during core reload. The location of the RVLIS pins is not known, therefore, it is assumed for this evaluation that the pins could reach the RCS.

The potential for damage to plant components is considered less than minimal based on their very small size. If they become lodged in a fuel assembly, there is the potential to cause fretting to the fuel rod cladding. However, the likelihood of fuel clad fretting and failure is low based on the geometry of the crimps and pins (they are not long, stringy or spiral in shape) and the limited number of crimps (five) and pins (two) that might find their way to the core.

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

ECs 618443, 618441 and 618648 evaluated the effects of the foreign material on RCS, ECCS and fuel components. The loose parts will not increase the frequency of occurrence of a stuck control rod, RCP locked rotor, RCP seal failure, RCS decreased flow, or cause a steam generator tube rupture. Since the likelihood of a accident is not increased and the components needed to mitigate the consequences to an accident are not more than minimally impacted, the proposed activity does not result in a more than minimal increase in the consequences of an accident previously evaluated in the UFSAR.

If the crimps and pins are transported to the RCS, the potential to damage various applicable components within the RCS and interconnecting systems was evaluated. It was concluded the foreign material does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. The introduction of the crimps and pins to the reactor may cause fretting of the fuel cladding, however UFSAR section 4.2 states dose limits given in 10CFR100 and 10CFR50.67 will not be exceeded due to a small number of rod failures. The likelihood

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron / Unit 0,1

Activity/Document Number: EC 618443 **Revision Number:** 0

Title: Lost SFP Crimps and RVLIS Pins

of multiple fuel rod failures is unlikely. Therefore the fuel, vessel, RCS, and connected systems and components would not malfunction in a manner different than has been previously evaluated in the UFSAR. Thus the consequences of a malfunction are not changed.

The loose parts are very small and would have minimal impact. The parts do not provide a means to initiate an accident of a different type than any previously evaluated in the UFSAR. The components have been evaluated in ECs 618441, 618443 and 618648 and it has been determined that they would continue to perform their design functions and any operational disturbance would be within the plant's design and operational basis. Thus the loose parts would not result in an SSC malfunctioning with a different result than what was previously evaluated in the UFSAR.

The three fission product barriers (fuel clad, RCS, containment) were considered. A fuel failure in itself will not exceed or alter a design basis limit for a fission product barrier (DBLFPB) a fuel failure in itself will not exceed or alter a design basis limit for a fission product barrier (DBLFPB) because per UFSAR section 4.2, dose limits given in 10CFR100 and 10CFR50.67 will not be exceeded due to a rod failure. Therefore, the loose parts do not result in a DBLFPB as described in the UFSAR being exceeded or altered.

The loose parts evaluation does not involve any evaluation methodologies used to establish design bases or safety analyses.

It is concluded the loose parts will not cause more than a minimal impact on plant components and systems.

Based on this evaluation, the proposed activity can be performed without prior NRC permission per the applicable governing procedures.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☐	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-17-001	**Rev.**	1

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Units 1 and 2

Activity/Document Number: EC 400957 / EC 402963 / DRP 17-015/ DRP 17-016 **Revision Number:** 000/ 000 /N/A/N/A

Title: Westinghouse Ovation Digital Upgrade for Rod Control Logic Cabinet (N-1 Outage)

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

The proposed activity (EC 400957 for Unit 1 and EC 402963 for Unit 2) replaces components within the rod control logic cabinets (1RD07J and 2RD07J) as part of an overall phased project to upgrade key non-safety-related process control systems, integrating them into an Ovation-based distributed control system. The upgrade is being performed in accordance with the recommendations of the original equipment manufacturer (Westinghouse). Separate activities (EC 400959 and EC 402965) are installing the Ovation distributed control system "infrastructure" (hardware and software) necessary to support the upgrade of the rod control logic cabinet. Separate activities are also installing an upgrade of the Ovation-based digital electro-hydraulic (DEH) turbine control system and connecting the upgraded DEH system to the DCS (EC 400958 and EC 402964). However, the turbine DEH control system will be connected to the DCS at the same time the upgrade of the logic cabinet is connected to the DCS.

This activity will replace the existing logic cabinet internal hardware and components. The majority of the new components will be furnished as a pre-fabricated panel assembly sized according to the dimensions of the existing logic cabinet, which will be screwed to the existing rails inside the cabinet. New components will consist of redundant controllers, input/output (I/O) modules, media converters, power supplies, line filters, circuit breakers, surge suppressors, a power distribution module, a transition panel which allows connecting to additional branches of I/O modules, a temperature and humidity transmitter, and terminal blocks.

The upgraded rod control logic cabinet is being designed by Westinghouse, the original designer and original equipment manufacturer, to reproduce the existing logic cabinet functions using new hardware and software. The rod speed and direction signals will be reproduced using new hardware and software. Instead of being adjustable via potentiometers at the cabinet, these speeds will be adjustable via the software server / engineering workstation. In auto mode, the rod speeds will be developed from the analog input using new hardware and software in the logic cabinet. Existing rod speeds and stop interlocks will be maintained. The potential for a single failure causing the rod speed to reach 77 steps/min will be eliminated, such that the maximum speed will be limited to 72 steps/min. The existing bank sequencing, bank overlap, and group alignment functions and the existing timing profile will be reproduced using new hardware and software. The bank overlap and timing profile will be adjustable via the software server / engineering workstation. As at present, overlap between successive banks will be adjustable between 0 and 50% of bank height. The upgraded rod control logic cabinet will rely on redundant controllers to perform functions that are at present performed by various cards. These functions include development of the timing profiles sent to the power cabinets, group sequencing, and bank overlap.

New internal cabinet cables will be installed that connect the Ovation I/O, bridge rectifier circuits and relays to the existing field terminal blocks located on the sides of the cabinet. The pulse-to-analog (P/A) converters in the DC hold cabinets (1RD08J and 2RD08J) will be removed and replaced with analog output signals originating in the logic cabinet.

To provide maximum fault tolerance for loss of power conditions, the upgraded cabinet will utilize three independent AC-power feeds. The existing power feeds provided to the cabinet from the motor-generator (MG) sets will be retained by this modification. A new third power feed will be provided from a spare circuit breaker in existing 120/240 distribution panels 1VE13E and 2VE13E. A new uninterruptible power supply will be installed in cabinets 1RD08J and 2RD08J to enhance the reliability of this power source. Auctioneering of the power sources will be provided by a new transfer switch, also installed in cabinets 1RD08J and 2RD08J. In the event that one of these supplies fails, the other power source will continue to provide power to the Ovation system, assuring normal operation of the system.

In the control room, the existing 100 VDC electromechanical rod group step counters, fifteen per unit, will be replaced with new 24 VDC digital step counters of the same form, fit, and function. The rod control step counters will be mounted in the existing step counter housings in panels 1PM05J and 2PM05J. The existing rod in and rod out lamps will also be replaced with new LEDs/lamps that are compatible with the 24 VDC circuit. Since the portion of the circuit that is being modified from 100 VDC to 24 VDC is also used by the loose parts monitoring system (LPMS), the LPMS circuitry will be modified to accommodate this change.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Units 1 and 2

Activity/Document Number: EC 400957 / EC 402963 / DRP 17-015/ DRP 17-016 **Revision Number:** 000/ 000 /N/A/N/A

Title: Westinghouse Ovation Digital Upgrade for Rod Control Logic Cabinet (N-1 Outage)

The rod bank position output signals to plant process computer (PPC) cabinet 1CX02J and 2CX02J will be abandoned. This data will now be provided to the PPC instead via the Modbus interface from Ovation to the PPC, installed as part of a separate activity (EC 400959 and EC 402965). The rod supervision PPC software will be revised by this activity to support rod bank position data being delivered to it via the Modbus interface to Ovation rather than the existing method of insert/withdraw pulses being read by the PPC I/O system. In addition, rod bank position information for group 1 as well as group 2 rod positions will be used in the program, whereas the existing program only uses group 1 data.

DRP 17-015 (Unit 1) and DRP 17-016 (Unit 2) will revise UFSAR Section 7.7.1.2.2 to incorporate changes associated with the Ovation-based control system upgrade.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

The upgrade of the rod control logic cabinet is one of several upgrades of non-safety-related process control systems being performed because the existing 7300-series control systems have exhibited performance, maintenance and reliability issues. It has been determined that the appropriate long term solution is to upgrade the existing 7300-series equipment to modern digital controls to enhance its reliability and overall lifespan.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

The basic operation of the rod control system will not be changed. The upgrade of the logic cabinet will involve minor changes to the operator interface with the rod control system (e.g., slight differences in the appearance of the step counter displays and in-out lights).

The functional and performance characteristics of the existing rod control logic cabinet will be maintained. Therefore, the proposed activity does not impact the design bases or safety analyses described in the UFSAR.

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

The proposed activity involves the replacement of components within the rod control logic cabinets (1RD07J and 2RD07J) as part of an overall phased project to upgrade key non-safety-related process control systems, integrating them into an Ovation-based distributed control system. The upgrade is being performed in accordance with the recommendations of the original equipment manufacturer (Westinghouse) to address equipment reliability and obsolescence issues. The upgrade was designed to match the existing functional and performance requirements of the rod control system.

Evaluations of the Ovation platform and its successful use in non-safety-related process control applications in the nuclear industry indicate that the system is expected to perform dependably. The application-specific software is being provided by Westinghouse – the original equipment supplier, who is employing their established product development processes used for the nuclear industry – to match the existing functional and performance characteristics of the rod control logic cabinet. The basic operation of the rod control system will not be changed, although the upgrade will involve minor changes to the operator interface with the rod control system. Therefore, the operator interface with the upgraded rod control system does not fundamentally change or adversely affect the methods used by the operator in performing or controlling a UFSAR-described design function. However, the upgraded rod control logic cabinet combines functions previously performed by multiple cards into redundant controllers, introducing the potential for failures to create new malfunctions which could adversely affect a UFSAR described design function. In addition, connecting the upgraded logic cabinet to a network linked to other plant control system introduces the potential for a software-based common cause failure that could adversely affect multiple systems. Therefore, 50.59 Evaluation 6G-17-002 was performed to address these issues.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Units 1 and 2

Activity/Document Number: EC 400957 / EC 402963 / DRP 17-015/ DRP 17-016 **Revision Number:** 000/ 000 /N/A/N/A

Title: Westinghouse Ovation Digital Upgrade for Rod Control Logic Cabinet (N-1 Outage)

The 50.59 Evaluation determined that the redundant hardware and improved diagnostics combined with the reliability of the new equipment and software and the fact that single failures remained bounded by existing single failures (as determined in a failure modes and effects analysis) did not result in a more than minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR or in a more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The 50.59 Evaluation also found that an analysis of the susceptibility of both the Ovation platform and the DCS application-specific design features to potential digital system failures – including software-based common cause failures – was performed using the guidance of EPRI Technical Report 3002005326 ("Methods for Assuring Safety and Dependability when Applying Digital Instrumentation and Control Systems"). In accordance with the EPRI guidance, the analysis considered the following potential sources of failure: random hardware failure, environmental disturbances, design defects, and operations or maintenance errors. Since individual device hardware failures had been previously addressed in a failure modes and effects analysis, the susceptibility analysis addressed hardware failures in shared resources. The analysis of design defects as a potential source of failure included a consideration of software-based common cause failures. The susceptibility analysis reviewed the design with respect to the various EPRI-identified preventive and limiting measures to determine whether these measure were sufficient to reduce the likelihood of a common cause failure to "Level 2", which is defined in the EPRI guidance as a failure likelihood at or below that of failures considered sufficiently unlikely that they would not typically be postulated and analyzed as part of a plant safety analysis report (e.g., random hardware failures or common cause failures of identical components caused by wear out and aging or by design or maintenance errors). The susceptibility analysis found that the design was adequate to reduce the likelihood of common cause failures in the Ovation platform or the specific DCS application to the desired level (i.e., EPRI Level 2).

On the basis of the susceptibility analysis, the 50.59 Evaluation concluded that the increase in the likelihood of accidents or malfunctions due to a software common cause failure is no more than minimal, and that the activity does not create the possibility of an accident or malfunction of a different type. The radiological consequences of previously evaluated accidents or malfunctions were not affected.

The same system parameters will continue to be monitored and alarmed with the rod control system upgrade, and the same decisions and actions will occur in the control of plant equipment and system during transients. The changes to the logic cabinet do not affect the overall operator response time and do not increase operator burden. The upgrade does not introduce any changes from manual to automatic initiation (or vice-versa) of functions. The upgrading of operator workstations does not fundamentally change the data presentation or the operator interface. As a result, the method by which the operators perform or control the logic cabinet design functions is not adversely affected. Thus the proposed activity does not involve a change to any procedure that affects how SSC design functions are performed or controlled.

The upgrade of the rod control logic cabinet in accordance with OEM recommendations does not involve a method of evaluation described, outlined, or summarized in the UFSAR that is used in the safety analyses or used to establish the design bases. Supporting analyses of electrical loading, structural loading, loop scaling and setpoints, and HVAC loading changes do not involve changes to any UFSAR-described methodology.

The proposed activity does not involve a test or experiment not described in the UFSAR. The upgrade of the logic cabinet system does not affect the Technical Specifications or the Facility Operating License.

Therefore, the proposed activity can be implemented per plant procedures without obtaining a license amendment.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Units 1 and 2

Activity/Document Number: EC 400957 / EC 402963 / DRP 17-015/ DRP 17-016 **Revision Number:** 000/ 000 /N/A/N/A

Title: Westinghouse Ovation Digital Upgrade for Rod Control Logic Cabinet (N-1 Outage)

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	6E-17-024	**Rev.**	0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-17-002	**Rev.**	0

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

50.59 REVIEW COVERSHEET FORM

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Station/Unit(s): Byron Unit 1&2

Activity/Document Number: EC 400958 / 402964 **Revision Number:** 000/000

Title: Westinghouse Ovation Digital Upgrade for DEH (N-1 Outage)

NOTE: For 50.59 Evaluations, infor NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

The proposed activity involves the upgrade of the Ovation-based digital electro-hydraulic (DEH) turbine distributed control system (DCS) in accordance with the recommendation of the original equipment manufacturer (Westinghouse). The existing OCR 161 controllers will be replaced with OCR 1100 controllers within the 1(2)PA22J DEH cabinet for integration with other Ovation DCS upgrades being implemented by other activities (EC 400957/EC402963 – "Westinghouse Digital Upgrade for Rod Control Logic Cabinet and EC 400959/EC 402965 – "Westinghouse Ovation Digital Upgrade for Infrastucture components"). This activity will also perform changes to network equipment within the DEH cabinet to further enhance system reliability. The operator workstations in the Main Control Room will be replaced. This activity also introduces a Heater Isolation Runback "soft" button feature as part of the DEH system user interface. The upgrade will also replace remote valve positioner modules for integration with the new OCR 1100 controllers to help eliminate failure modes and effects in turbine stop and control valve positioning components that have been identified in the industry in earlier Ovation systems. Minor changes will be made to the power supplies in the generator stator cooling panel 1(2)GC01J.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

The turbine controls were previously upgraded to a DEH system which uses the Ovation platform. The proposed activity involves a second upgrade to make the DEH DCS compatible with upgrades of other plant distributed control systems to a more recent Ovation platform (under separate activities referenced above). The proposed heater isolation runback feature has been requested by Operations, as part of the resolution to an INPO area for improvement (AFI), to improve the plant response to a "HI-2" level alarm in a 15, 16, 25, or 26 feedwater heater.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

The proposed activity involves the replacement and reconfiguration of select components in the Ovation-based DEH system to make that system compatible with a later version of the Ovation platform that will be used for upgrading the DEH and other non-safety-related process control systems and incorporating them into an Ovation-based distributed DCS. The proposed activity also includes additional minor upgrades to the DEH DCS. The upgrade will involve slight changes to the operator interface with the DEH system (e.g., newer workstations). The new runback feature, which will ramp the turbine power down at 100 MW/Min until an operating level of 1050 MWe is reached, is intended to assist the operators in rapidly reducing power following receipt of a HI-2 level alarm for the 15, 16, 25, or 26 feedwater heaters in order to prevent a reactor trip due to the loss of feedwater heating. The net decrease from full power has been selected to bound the power reduction required once the cascading effects of heater shell-side isolation due to the HI-2 alarm actuation are taken into account, as specified in BOP HD-6T1. The new runback feature is similar to existing runback features for the loss of a main feedwater or condensate/booster pump.

The functional and performance characteristics of the existing turbine control system will be maintained. Therefore, the proposed activity does not impact plant operations, design bases or safety analyses described in the UFSAR.

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

The proposed activity involves the replacement and reconfiguration of select components in the Ovation-based DEH DCS, in accordance with the recommendations of the OEM (Westinghouse), in order to incorporate that system into an Ovation-based DCS that will be used for the DEH and other non-safety-related process control systems. The overall Ovation-based DEH control system architecture and its capabilities are being maintained, and the upgrade of the DEH system will reproduce the existing

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Station/Unit(s): Byron Unit 1&2

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Title: Westinghouse Ovation Digital Upgrade for DEH (N-1 Outage)

turbine control system functional and performance characteristics using the new controllers, workstations, and associated software. Evaluations of the Ovation platform and its successful use in non-safety-related process control applications in the nuclear industry indicate that the minor upgrades to the DEH system will not affect its dependability. The application-specific software is being provided by Westinghouse – the original equipment supplier, who is employing their established product development processes used for the nuclear industry – to match the existing functional and performance characteristics of the DEH system. As a result, the modified Ovation platform and the upgraded DEH controls are expected to perform dependably. However, connecting the DEH system to a network linked to other plant control systems introduces the potential for a software-based common cause failure that could adversely affect multiple systems. This was considered a change to an SSC that adversely affects a UFSAR-described design function; therefore, a 50.59 evaluation 6G-17-003 was performed.

The 50.59 Evaluation found that an analysis of the susceptibility of both the Ovation platform and the DCS application-specific design features to potential digital system failures – including software-based common cause failures – was performed using the guidance of EPRI Technical Report 3002005326 ("Methods for Assuring Safety and Dependability when Applying Digital Instrumentation and Control Systems"). In accordance with the EPRI guidance, the analysis considered the following potential sources of failure: random hardware failure, environmental disturbances, design defects, and operations or maintenance errors. Since individual device hardware failures had been previously addressed in a failure modes and effects analysis, the susceptibility analysis addressed hardware failures in shared resources. The analysis of design defects as a potential source of failure included a consideration of software-based common cause failures. The susceptibility analysis reviewed the design with respect to the various EPRI-identified preventive and limiting measures to determine whether these measures were sufficient to reduce the likelihood of a common cause failure to "Level 2". Level 2 is defined in the EPRI guidance as a failure likelihood at or below that of failures considered sufficiently unlikely that they would not typically be postulated and analyzed as part of a plant safety analysis report (e.g., random hardware failures or common cause failures of identical components caused by wear out and aging or by design or maintenance errors). The susceptibility analysis found that the design was adequate to reduce the likelihood of common cause failures in the Ovation platform or the specific DCS application to the desired level (i.e., EPRI Level 2).

On the basis of the susceptibility analysis, the 50.59 Evaluation concluded that the increase in the likelihood of accidents or malfunctions due to a software common cause failure is no more than minimal, and that the activity does not create the possibility of an accident or malfunction of a different type. The radiological consequences of previously evaluated accidents or malfunctions were not affected.

The minor changes to the operator interface with the upgraded DEH system do not involve a change to a procedure that adversely affects how UFSAR-described SSC design functions are performed or controlled. The new heater isolation runback feature is similar to existing turbine runback features, which allow for a rapid reduction in turbine and reactor power to prevent an upset condition from leading to a reactor trip, while remaining well within the load rejection capability of the reactor control and steam dump control systems. Therefore, this aspect of the proposed activity does not adversely affect a UFSAR-described design function. The runback on high-high heater level simplifies and consolidates existing actions by providing a single button with a pre-programmed ramp rate and power level to bound the power reduction required once the cascading effects of heater shell-side isolation are taken into account. The new runback feature will remain a manual action but will reduce the operator burden, with no new potential failure modes in the interaction of operators with the plant systems. Therefore, in accordance with the guidance provided in Section 5.2.2.2 of LS-AA-104-1000 (50.59 Resource Manual), this aspect of the proposed activity does not involve a change to a procedure that adversely affects the method of performing or controlling a UFSAR-described design function.

The upgrade of the DEH DCS in accordance with OEM recommendations and the introduction of the new runback feature do not involve a method of evaluation described, outlined, or summarized in the UFSAR that is used in the safety analyses or used to establish the design bases. Additionally, supporting analyses of electrical power loading, heat loading, HVAC loading changes and equipment seismic qualification do not involve any changes to UFSAR described methodologies. The proposed activity does not involve a test or experiment not described in the UFSAR. The upgrade of the DEH system does not affect the Technical Specifications or the Facility Operating License.

Therefore, NRC approval is not required for the proposed activity. The activity may be implemented per the applicable governing procedure.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 1&2

Activity/Document Number: EC 400958 / 402964 **Revision Number:** 000/000

Title: Westinghouse Ovation Digital Upgrade for DEH (N-1 Outage)

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	6E-17-025	**Rev.**	0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-17-003	**Rev.**	0

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): **Byron / Unit 1**

Activity/Document Number: **EC # 619975** **Revision Number:** **0**

Title: TCCP to Install Pneumatic Jumper to Open 1WG074

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

Temporary Configuration Change Package (TCCP) EC 619975 allows the temporary installation of a pneumatic jumper to allow opening of 1WG074 despite the associated solenoid valve, which is intended to automatically close 1WG074 when the Feedwater Pump Seal Water Collection Tank level is low, being de-energized due to a loss of the associated control power.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

IR 4018759 documents that MCC 133Z2 unexpectedly lost power due to the MCC feed breaker (1AP18E-3C) tripping open. As a result of this MCC being lost, a number of components failed to the de-energized state. A solenoid valve associated with 1WG074, which is normally powered from MCC 133Z2, causes 1WG074 to fail closed when the solenoid is de-energized. Closure of 1WG074 isolates the outlet from the Feedwater Pump Seal Water Collection Tank (1WG02T).

IR 4018774 documents that a pneumatic jumper was installed as an immediate action to place the situation in a safe and stable condition per PI-AA-120 to allow opening of 1WG074 to restore flow to the Unit 1 Condenser, which stops overflow from the Unit 1 Feedwater Pump Seal Water Collection Tank to the Turbine Building Floor and thereby conserves Condensate Storage Tank water. The installation of this jumper is a TCC and requires documentation per CC-AA-112.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

This activity restores the ability of 1WG074 to open and allow level control of the Unit 1 Feedwater Pump Seal Water Collection Tank by bypassing the associated solenoid valve that automatically causes 1WG074 to close upon a Low Level condition in the tank. This activity modifies the non-safety related control loop to allow continued operation despite the loss of power to one component within the control loop. The level control loop normally functions to automatically control tank level using the output of a level controller to proportionally modulate the valve position of 1WG074. The level control loop also receives a signal from a redundant level switch, which functions to close 1WG074 if the redundant level switch senses a low level condition due to a malfunction of the normal level controller. This TCCP will bypass the level control loop and tank level will be manually controlled by an operator continuously stationed at the tank. The automatic isolation upon a low level alarm from the redundant level switch for the associated tank is also defeated as a result of this TCCP.

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

The UFSAR does not describe or credit the Feedwater Pump Seal Water Collection Tank Outlet Valve (1WG074) or the associated Feedwater Pump Seal Water Collection Tank level control loop or its function. The automatic isolation function prevents an air inleakage path to the condenser from being established in the event that the tank is completely emptied. Loss of condenser vacuum and turbine trip are listed as initiating events for decrease in heat removal by the secondary system transients in UFSAR Section 15.2. Per UFSAR 15.2.3, this is a ANS Condition II transient of moderate frequency. However, the condenser has design features to maintain vacuum such as the SJAEs and vacuum hogging pumps. Existing station procedures direct operators to take action to maintain condenser vacuum. In the event of inleakage through the seal water collection tank, operators would isolate 1WG033 or 1WG035. These design features and proceduralized actions ensure that condenser vacuum is maintained. Thus, defeating the automatic isolation feature of 1WG074 would not significantly affect the frequency of loss of condenser vacuum. The ANS Condition II classification of the transient is not affected because the frequency of occurrence is still moderate. Therefore, the proposed activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

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Station/Unit(s): Byron / Unit 1

Activity/Document Number: EC # 619975 **Revision Number:** 0

Title: TCCP to Install Pneumatic Jumper to Open 1WG074

The collection tank and level control system are not described in the UFSAR. The condenser is described, but is not credited with mitigating the consequences of an accident or performing any other safety function. No other SSCs are affected by the proposed activity. Therefore, the proposed activity does not increase the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

While loss of condenser vacuum is described as an initiating event, initiating events do not affect the consequence of an accident. Furthermore, the condenser is not credited with performing a safety function. Therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

While loss of condenser vacuum is listed as an initiating event in UFSAR 15.2, the consequence of a loss of condenser vacuum and its effect on SSCs important to safety is not changed. Therefore, the proposed activity does not result in an increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Decrease in heat removal by the secondary is the only transient that could be initiated by a loss of condenser vacuum. Therefore, the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

Malfunction of 1WG074 could potentially result in air inleakage to the condenser. Air inleakage and/or loss of condenser vacuum have already been evaluated in the UFSAR. Therefore, the proposed activity does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

Neither the FW pump seal water collection tank level control system nor the condenser are fission product barriers and do not interface with any fission product barriers either directly or indirectly. Therefore, the proposed activity does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

The Unit 1 Feedwater Pump Seal Water Collection Tank and associated level control equipment is not associated with any element of UFSAR described evaluation methodology used in establishing the design bases or used in the safety analyses. The proposed activity does not create a new or alternative evaluation methodology. Loss of condenser vacuum is an initiating event for, but does not affect the safety analyses for the decrease in heat removal by the secondary system transient. Therefore, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

The Feedwater Pump Seal Water Collection Tank and associated level control equipment are not addressed in the Technical Specifications or in the Operating License. Therefore, the Technical Specifications and Operating License do not require change.

Based on this evaluation, the proposed activity does not require prior NRC approval for implementation.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☐	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-17-004	**Rev.**	0

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 617266 / DRP 17-023 **Revision Number:** 000 / 0

Title: Provide Capability to Disable Water Hammer Prevention System (WHPS) Feedwater (FW) Isolation Signals During Normal Power Operation & Change WHPS SG Low Pressure/Level Control Relay Operation From Actuate On De-Energized To Actuate On Energized

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

This activity will modify the Non-Safety Related Feedwater System (FW) Water Hammer Prevention System (WHPS). Implementation of this activity encompasses the following.

(1) Provide the capability to disable the WHPS Steam Generator low level and low pressure feedwater isolation signals during normal operation at power levels that require the Main FW Isolation Valves 2FW009A/B/C/D to be open.

The proposed change to disable the WHPS FW isolation involves installing a new switch in the circuit between two terminal block points for each of the Feedwater Isolation Valves, 2FW009A/B/C/D, and FW Preheater Bypass Valves, 2FW039A/B/C/D, and repositioning the control switch from "Auto" to "Open" for each of the FW Tempering Isolation Valves 2FW035A/B/C/D valves. The associated annunciator circuit and annunciator tile for 2FW035A/B/C/D position is modified to reflect the new normal position for power operation (above approximately 25%). The WHPS FW isolation signals to the FW Isolation Bypass Isolation Valves 2FW043A/B/C/D valves are not disabled (these valves are normally in the closed position above 25% power level).

The following changes will be made to plant operating procedures to implement manual enabling/disabling of the WHPS FW isolation auto close signals:

- 2BGP 100-1, PLANT HEATUP, will be revised to verify that WHPS FW isolation control circuits are enabled prior to starting any FW pumps.
- 2BGP 100-3, POWER ASCENSION, will be revised to disable WHPS FW isolation after the 2FW009A/B/C/D valves are open.
- 2BGP 100-4, POWER DESCENSION, will be revised to enable WHPS FW isolation control circuits for the 2FW035A/B/C/D and 2FW039A/B/C/D valves prior to closing the 2FW009A/B/C/D valves, and then perform a follow-up action to enable WHPS FW isolation control circuits on the 2FW009A/B/C/D valves.
- 2BGP 100-4T4, REACTOR TRIP POST RESPONSE GUIDELINE, will be revised to verify that the 2FW009A/B/C/D, 2FW043A/B/C/D, and 2FW039A/B/C/D valves are closed prior to starting the Startup Feedwater Pump. Action will also be taken to enable WHPS FW isolation control circuits later in the post trip response.

UFSAR Section 10.4.7.3 will be revised as a result of this activity to describe the capability to disable the WHPS FW isolation control circuits when the Engineered Safety Features Actuation System (ESFAS) Feedwater Isolation feature is active.

(2) Change the logic of the WHPS Steam Generator (SG) low level and low pressure control relays such that, when WHPS is enabled, the WHPS actuation will occur when the relays are energized (versus currently de-energized).

As presently configured, the Non-Safety Related Feedwater (FW) Water Hammer Prevention System (WHPS) causes closure of the Feedwater Isolation Valves (FWIVs) when control relays "de-energize". The proposed activity will change this logic such that closure of the FWIVs will occur when the control relays are "energized".

Implementation of this activity requires swapping wires from nomally open (NO) contacts to the normally closed (NC) contacts on existing WHPS relays that actuate on Steam Generator (SG) Level Low and SG Pressure Low conditions. Also, the existing Time Delay Relays related to WHPS SG Level Low Time Delay that currently start on a "de-energized" condition, will be replaced with new Time Delay Relays that start on an "energized" condition.

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Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 617266 / DRP 17-023 **Revision Number:** 000 / 0

Title: Provide Capability to Disable Water Hammer Prevention System (WHPS) Feedwater (FW) Isolation Signals During Normal Power Operation & Change WHPS SG Low Pressure/Level Control Relay Operation From Actuate On De-Energized To Actuate On Energized

Additionally, this activity will functionally remove a second/redundant relay in the WHPS steam generator low pressure and low level permissive circuit for the FWIVs (2FW009A/B/C/D) that was added in 2005 under EC 348112 to eliminate a single point vulnerability the WHPS circuits caused by "de-energization" of the single relay in the original design. Upon installation of this EC 617266, WHPS actuation via this second/redundant relay is no longer needed since the previous single point vulnerability (WHPS actuation on "de-energized" condition) will no longer exist.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

This activity is being initiated following an unexpected loss of power to Non-Safety Related MCC 234V4, which subsequently caused loss of power to the WHPS control relays in cabinet 2PA28J related to 2B/2C SG Level Low and SG Pressure Low conditions. When the relays changed state from "energized" to "de-energized" due to the loss of power, the WHPS generated a signal to automatically close the 2B/2C FWIVs resulting in decreasing 2B/2C SG levels. In response to the inadvertent feedwater isolation and decreasing SG levels, operators manually tripped the reactor.

Implementation of the proposed change to provide the capability to disable the WHPS Steam Generator low level and low pressure FW isolation control circuits during power operation above start-up levels, will address an adverse vulnerability which can cause spurious/unnecessary WHPS actuation, feedwater isolation, loss of normal feedwater flow, and reactor trip solely due to the loss of a single non-safety related power supply for the relays. The changes to the plant operating procedures are made to control when the WHPS FW isolation circuits are disabled/enabled.

Implementation of the proposed change to the control relay logic of the WHPS Steam Generator (SG) low level and low pressure control relays such that WHPS actuation will occur when the relays are energized (versus currently de-energized), will prevent similar reactor trips in the future when WHPS is enabled.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

(1) The safety function of the Feedwater Isolation Valves is to close, isolating main feedwater flow upon a safety injection signal, feedwater isolation signal and high-2 SG level. As described in UFSAR section 10.4.7.3, "Several water hammer prevention features have been designed into the feedwater system. These features are provided to minimize the possibility of various water hammer phenomena in the steam generator preheater, steam generator main feedwater inlet piping, and the steam generator upper nozzle feedwater piping." Steam generator (two-out-of-three logic) low level trips are provided to close all feedwater isolation valves (2FW009A/B/C/D), feedwater isolation bypass valves (2FW043A/B/C/D) and feedwater preheater bypass valves (2FW039A/B/C/D). Steam generator (two-out-of-three logic) low pressure trips are provided to close all feedwater isolation valves (2FW009A/B/C/D), feedwater isolation bypass valves (2FW043A/B/C/D), feedwater preheater bypass valves (2FW039A/B/C/D), and the feedwater bypass tempering valves (2FW035A/B/C/D).

The UFSAR described water hammer prevention features were included in the NRC's basis for approval of (1) the designs of the Steam Generator and the FW/AFW systems, and the measures to address Unresolved Safety Issue (USI) A-1, Steam Generator Water Hammer; and (2) the removal of pipe restraints resulting from the elimination of arbitrary intermediate breaks in the FW system piping system design.

The purpose of the WHPS FW isolation is to prevent water hammer by precluding the admission of cold FW into a steam-voided preheater section of the steam generator (when SG narrow range level is less than 5% or when SG pressure is less than 600 psig). When the WHPS FW isolation is disabled at power the water hammer prevention function will be fulfilled by the safety related ESFAS signals function to close the feedwater isolation valves on low steam line pressure safety injection or steam generator water level low-low reactor trip coincident with Tave less than 564° F. Note that the 2FW039 valves close on reactor trip and no coincident Tave is required. At low power levels or during startup the WHPS FW isolation features will be enabled. The design bases function to preclude the admission of cold FW into a steam-voided preheater section of the steam generator is maintained.

The other UFSAR described WHPS features will not be altered by this activity. Operating procedures associated with only providing feedwater to the upper nozzle of the steam generator during startup and low load conditions are unchanged. Temperature monitoring and alarms to detect possible back leakage of steam from the steam generators into the feedwater piping are unchanged. Forward flushing and temperature permissive interlocks associated with opening the FWIV are also unchanged. This activity does not alter split feedwater flow operation.

Implementation of this activity does not impact the safety related circuits described in the UFSAR and Technical Specifications that generate an independent Feedwater Isolation Signal, including Safety Injection (SI), SG Level HI-2 Trip (P-14), and Reactor Trip (P-4) coincident with RCS Average Temperature Low at 564°F. Additionally, implementation of this activity does not impact the Reactor Trip signal generated on low-low steam generator water level

Implementation of this activity will require new operator actions to appropriately enable/disable WHPS FW isolation in the plant startup, power ascension, power descension, and plant trip response procedures. Operators will reposition the control board hand switch for the 2FW035 A/B/C/D valves and reposition the switches for the 2FW009A/B/C/D and 2FW039A/B/C/D valve circuits.

- For startup WHPS FW isolation will be verified to be enabled prior to starting any FW pumps. Once the 2FW009 valves are open, action will be taken to disable WHPS FW isolation for the 2FW009, 2FW035, and 2FW039 valves.
- For a normal shutdown or low power operation action will be taken to re-enable WHPS FW isolation for the 2FW035 and 2FW039 valves prior to closing the FW009 valves. WHPS FW isolation for the 2FW009 valves will be enabled prior to defeating the ESFAS FW isolation signals.
- On a plant trip action will be taken to verify that the 2FW009, 2FW039, and 2FW043 valves are closed prior to starting the Startup FW pump. The 2FW035 valve flow path is used to direct FW to the upper nozzle. The UFSAR described main FW flow and temperature permissive interlocks will prevent inadvertent introduction of cold FW to the lower nozzle until action is taken to enable WHPS FW isolation.

There is no impact on any emergency operating procedures. The proposed operator actions are not time critical and do not impose a significant new burden on the operating crew.

(2) This activity does not impact the independent Safety Related circuits described in the UFSAR and Technical Specifications that also generate a Feedwater Isolation Signal, including Safety Injection (SI), SG Level HI-2 Trip (P-14), and Reactor Trip (P-4) coincident with Low RCS Average Temperature at 564°F.

The present logic configuration can cause feedwater isolation even though the SG Level and SG Pressure conditions necessary for WHPS actuation do not exist. Modifying the WHPS logic will prevent spurious actuation when WHPS is enabled and prevent a unit trip merely as a result of loss-of-power to control relays. The existing configuration causes the WHPS to actuate on loss of Non-Safety Related power to the control relays. This power fail-safe design is not required for the WHPS in the same manner as that required for the independent Safety Related protection circuits stated above. There are no UFSAR or design requirements for the WHPS to actuate due to a loss of power condition in this manner. The existing design has demonstrated a vulnerability that results in a significant plant transient and reactor trip. The modified design to change the logic of the WHPS Steam Generator (SG) low level and low pressure control relays such that WHPS actuation will occur when the relays are energized (versus currently de-energized), will eliminate this vulnerability.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 617266 / DRP 17-023 **Revision Number:** 000 / 0

Title: Provide Capability to Disable Water Hammer Prevention System (WHPS) Feedwater (FW) Isolation Signals During Normal Power Operation & Change WHPS SG Low Pressure/Level Control Relay Operation From Actuate On De-Energized To Actuate On Energized

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

50.59 Screening

WHPS Disable Capability; The UFSAR described water hammer prevention features were included in the NRC's basis for approval of (1) the designs of the Steam Generator and the FW/AFW systems, and the measures to address Unresolved Safety Issue (USI) A-1, Steam Generator Water Hammer; and (2) the removal of pipe restraints resulting from the elimination of arbitrary intermediate breaks in the FW system piping system design. As such, implementation of the capability to disable the WHPS Steam Generator low level and low pressure feedwater isolation signals is considered adverse and will be reviewed in the 50.59 evaluation.

WHPS Control Logic Change; The present logic configuration can cause feedwater isolation even though the SG Level and SG Pressure conditions necessary for WHPS actuation do not exist. Modifying the WHPS logic will prevent spurious actuation and prevent a unit trip merely as a result of loss-of-power to control relays. The modified logic will continue to provide the same Non-Safety Related WHPS function as described in the UFSAR. This function of the WHPS is Non-Safety Related and is independent of the Safety Related Engineering Safety Features (ESF) functions that also isolate feedwater. The WHPS system does not have a UFSAR described design function to isolate feedwater under any condition other than the low SG Level and low SG pressure conditions as described above. Additionally, the UFSAR does not discuss whether the WHPS is de-energize or energize to actuate. As presently configured, during a loss of power while at normal operation, the WHPS system will lose power and the relay logic will isolate feedwater from the SGs. This modification will eliminate this response on loss of power. However, a reactor trip on low SG level will occur prior to reaching the WHPS conditions of <5% SG Level and a Safety Injection actuation will occur on low SG pressure prior to reaching the WHPS conditions of <600 psig SG pressure. A reactor trip concurrent with low RCS Average Temperature or a Safety Injection actuation will close all FWIVs as required, independent of the WHPS. The independent feedwater isolation signals that occur following loss of power are safety-related. This activity does not impact the UFSAR described design function. The only circuit being altered is the FW WHPS circuit and the new design will function in the UFSAR described manner. The solenoids associated with the FWIVs are unchanged and will remain "Energize to Close", such that system reliability is not impacted. Modification testing will ensure that the FWIV closure circuit design function remains as described in the UFSAR. The devices being added (new time delay relays) meet the same design, performance and installation standards as the existing devices in the circuit. All 1E to non-1E circuit interfaces remain intact as well as train/non-train separation requirements as described in the UFSAR. In summary, this activity has been reviewed and determined to not have any adverse impacts to an UFSAR described design function.

**

WHPS Disable Capability; Implementation of this activity will require new operator actions to appropriately enable/disable WHPS FW isolation in the plant startup, power ascension, power descension, and plant trip response procedures. This change is considered adverse and will be reviewed in the 50.59 evaluation.

WHPS Control Logic Change; There are no procedure impacts with this change as the solenoids associated with the FWIVs are unchanged and will remain "Energize to Close". The UFSAR does not describe the logic mechanism for how the WHPS signals are sent to valves. As such, this change is not considered adverse. In summary, this activity has been reviewed and determined to not involve a change to a procedure that adversely affects how UFSAR described SSC design functions is performed or controlled.

**

There are no evaluation methodologies or alternative evaluation methodologies involved in this activity. This activity modifies the Unit 2 WHPS low SG level and low SG pressure FW isolation circuits to provide a means to disable the auto close signals when the unit is operating above approximately 25% power. This activity does not involve a method of evaluation as defined in LS-AA-104-1000. This activity will also modify the WHPS control logic so that WHPS actuation will occur when the relays for SG Level Low or SG Pressure Low are energized (versus currently de-energized), thus alleviating the potential for an inadvertent FWIV closure signal. Feedwater isolation from the WHPS is not related to any

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 617266 / DRP 17-023 **Revision Number:** 000 / 0

Title: **Provide Capability to Disable Water Hammer Prevention System (WHPS) Feedwater (FW) Isolation Signals During Normal Power Operation & Change WHPS SG Low Pressure/Level Control Relay Operation From Actuate On De-Energized To Actuate On Energized**

UFSAR described evaluation methodology. The logic mechanism for how the Non-Safety Related Feedwater WHPS signals are sent to valves requiring closure is not specified or referenced in the UFSAR and is not related to any UFSAR described evaluation methodology or used in establishing the design bases or safety analyses. Implementation of this activity does not impact the independent Safety Related Engineering Safety Features (ESF) FW isolation signals (Steam line pressure SI or the Reactor trip + Lo Tave) that also isolates feedwater. Therefore, this activity does not involve an adverse change to an element of a UFSAR described evaluation methodology, or use of an alternative evaluation methodology, that is used in establishing the design bases or used in the safety analyses.
**

The FW WHPS remains equivalent in function when feedwater isolation is appropriate on Low SG Level and Low SG Pressure. The revised design does not impact or interact with any other plant design functions. This activity requires post modification installation testing. Per LS-AA-104-1000, Section 4.2.2, Post-modification testing of approved facility changes is indistinguishable, in terms of their risk impact on the plant, from maintenance activities that restore SSCs to their as-designed condition. As such, installation and testing of approved facility changes are maintenance activities that must be assessed and managed in accordance with 10CFR50.65(a)(4), and therefore, this is not a test or experiment. The installation and testing activities will be structured to preclude interaction with other plant equipment and functions. Therefore, this activity does not involve a test or experiment not described in the UFSAR, where an SSC is controlled in a manner outside the reference bounds of the design.
**

The Non-Safety Related WHPS is not described in the Technical Specifications or Operating License and does not support the bases for any Technical Specification. The Technical Specifications Section 3.3.2 Engineering Safety Features (ESF) functions of the Feedwater Isolation Valves to close is not being changed in any way nor impacted by this activity. Therefore, this activity has no impact on the Technical Specifications or Operating License and no Technical Specification or Operating License changes are required.

50.59 Evaluation

Disabling the WHPS low SG level and low SG pressure FW isolation auto close signals above 25% power does not introduce the possibility of a change in the frequency of an accident because WHPS is not an initiator of any accident and no new failure modes are introduced. Disabling the WHPS low SG level and low SG pressure FW isolation auto close signals above 25% power reduces the probability of a loss of feedwater event due to a spurious WHPS actuation.

In the plant operating conditions under which the WHPS automatic FW isolation feature may be procedurally disabled (power level above approximately 25% power), the safety related ESFAS signals function to close the feedwater isolation valves on a low steam line pressure safety injection (< 640 psig) or steam generator water level low-low reactor trip (≤ 36.3% NR) coincident with low Tave (< 564° F). Note that the 2FW039A/B/C/D valves close on reactor trip and no coincident Tave is required. A review of the postulated transients that could create conditions for a water hammer event when power is above 25% was performed (Refer to EC 617266). The review determined that the ESFAS FW isolation signals (Steam line pressure SI or the Reactor trip + Lo Tave) will function to prevent water hammer by precluding the admission of cold FW into a steam- voided preheater section of the steam generator during power operating conditions. Therefore disabling the WHPS low SG level and low SG pressure FW isolation auto close signals above 25% power does not increase the probability of water hammer in the SG preheater and the main FW inlet piping. Thus the proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident caused by a water hammer event in the FW line or the Steam Generator (FW line break or steam generator tube rupture).

The defeated WHPS FW isolation function is not used or credited to mitigate any design basis event, analysis, or Emergency Operating Procedure Operator Action. The credited automatic function is the ESFAS FW Isolation which is a completely separate circuit. Thus the ESFAS FW Isolation is not affected by this change.
**

The change adds new operator actions to disable and enable the WHPS low level and low pressure FW isolation auto close signals. Failure to enable the WHPS low SG level and low SG pressure auto close signals for startup, low load, or increasing load conditions could potentially result in a malfunction of equipment important to safety, specifically prevention of water hammer in the steam generator preheater and/or steam generator main feedwater (FW) inlet piping. The likelihood of occurrence of a malfunction associated with the new operator actions is not more than minimal based on the following:

- The actions to disable/enable the WHPS low SG level and low SG pressure FW isolation auto close signals has been added to the appropriate plant procedures and operators will be trained on the revised procedures.
- The actions to disable/enable the WHPS low SG level and low SG pressure FW isolation auto close signals do not have a required time for completion. Above 25% power there is no specific time requirement for disabling the WHPS low SG level and low SG pressure FW isolation auto close signals. During normal power descension there are no time limitations to enable WHPS, as this is directed by procedure prior to manually blocking SG steamline low pressure Safety Injection during shutdown/cooldown. There is no impact on any emergency operating procedures. The proposed operator actions for WHPS enable/disable during power ascension, power descension, and reactor trip are not time critical and do not impose a significant new burden on the operating crew.
- This change does not introduce credible errors during performance of the operator actions required for this change. Three actions are required to disable or enable WHPS following implementation of this activity.
  - Operating a single (new) switch for each 2FW009A/B/C/D valve circuit at the Auxiliary Safeguards Relay Cabinet, 2PA27 (Train A) or 2PA28J (Train B). Manipulation of the switch is considered a routine action with no credible failure risk. Opening/closing the switch does not initiate a water hammer prevention action; it disables/enables the respective WHPS circuit to the associated valve solenoid which remains unchanged by this activity as "Energize to Close", such that system reliability is not impacted.
  - Operating a single (new) switch for each 2FW039A/B/C/D valve circuit at the Auxiliary Safeguards Relay Cabinet, 2PA27 (Train A) or 2PA28J (Train B). These switches will be wired across the "original" water hammer prevention actuation relay contacts in cabinets 2PA27 (Train A) or 2PA28J (Train B), thereby allowing the capability to disable/enable the "original" contact actuation for the WHPS circuit to the associated valve solenoid which remains unchanged by this activity as "Energize to Open", such that system reliability is not impacted. Manipulation of the switch is considered a routine action with no credible failure risk.
  - Operation of the existing Main Control Room control switch (2HS-FW261/262/263/264) for the 2FW035A/B/C/D valves. With implementation of this activity, the 2FW035A/B/C/D valves will be kept open by placing the valve control switches to the OPEN position, thus bypassing the water hammer prevention circuit. When this control switch is placed in the AUTO position the 2FW035A/B/C/D valves will automatically close when there is a Steam Generator Low Pressure (2 out of 3) condition. This is a routine method of configuration control by operators. The associated valve solenoid remains unchanged by this activity as "Energize to Open", such that system reliability is not impacted.

  Operation of the "new" switches for the 2FW009A/B/C/D and 2FW039A/B/C/D valve circuits, and operation of the "existing" Main Control Room control switch (2HS-FW261/262/263/264) for the 2FW035A/B/C/D valves will be controlled as part of an approved procedure implemented as part of this activity. The switches to be installed and used, and hand switches to be manipulated per this activity will be clearly labelled per the Configuration Change to allow for clear and easy operator identification. Operation of the switches does not initiate a water hammer prevention actuation; therefore, there would be time to recover from unlikely actions such as blocking or enabling WHPs out of the expected sequence.
- The purpose of the WHPS FW isolation signals is to prevent water hammer by precluding the admission of cold FW into a steam-voided preheater section of the steam generator. A review of the postulated transients that could create conditions for a water hammer event when power is above 25% was performed (Refer to EC 617266). The review determined that the ESFAS FW isolation signals (Steam line pressure SI or the Reactor trip + Lo Tave) will function to prevent water hammer by precluding the admission of cold FW into a steam-voided preheater section of the steam generator during power operating conditions. Thus the system function to preclude the admission of cold FW into a steam-voided preheater section of the steam

generator is maintained.

When the WHPS FW isolation closure signals are disabled, the UFSAR described automatic action to isolate feedwater to the preheater region of the Unit 2 SGs on low SG level or low SG pressure will be fulfilled by the safety related ESFAS FW isolation signals. Appropriate administrative controls will be put in place to enable the automatic WHPS FW isolation function during plant startup, power descension, and after a plant trip when the ESFAS FW isolation is not available. Thus the proposed change does not permanently substitute manual action for automatic action for performing a UFSAR-described design function.

The WHPS low SG level and low SG pressure FW isolation auto close signals are classified as non-safety related. The WHPS FW isolation auto close signal for each individual SG is based on a two-out-of-three logic. The safety related ESFAS FW isolation signal on low steam line pressure is based on two-out-of-three logic in any one of the four steam lines. A reactor trip is actuated on two-out-of-four low-low SG water level signals occurring in any steam generator. A FW isolation signal is generated with the reactor trip coincident with a two-out-of-four Lo Tave signals. The safety related ESFAS FW isolation signals provide the same level of redundancy as the WHPS FW isolation signals. The safety related ESFAS components are considered to be more reliable than the non-safety related WHPS components. The highest risk for water hammer events is during startup and low power operation when the ESFAS FW isolation signals may be bypassed and only the WHPS FW isolation signals would be available for water hammer protection. This same level of redundancy will be in place with only the ESFAS FW isolation signals providing protection when the WHPS low SG level and low SG pressure auto close FW isolation signals are disabled.

The new switches used to disable water hammer protection on the 2FW009A/B/C/D and 2FW39A/B/C/D valves meet or exceed the quality requirements of the circuits they are installed in and are installed in accordance with approved quality controlled procedures to ensure the appropriate level of reliability. The switches for the 2FW009A/B/C/D valves are installed in series with the applicable contacts with lugs on the terminal block and are considered to be as reliable as the existing circuit. The switches on the 2FW039A/B/C/D valves are installed in parallel with the applicable contacts with lugs on the terminal block and are considered to be as reliable as the existing circuit. Existing procedures for Power Ascension (2BGP 100-3), Power Descension (2BGP 100-4) and Low Power Turbine Trip (2BOA TG-8) provide the necessary guidance for operator alignment of the FW system, prior to potential actuation of the WHPS, to prevent potential water hammer events. Byron Emergency Response Procedures (e.g. 2BEP ES-0.1, Reactor Trip Response) already provide the necessary guidance for restoration of water flow to the upper SG nozzle and isolation of the main FW nozzle and preheater section per the water hammer prevention guidance provided by Westinghouse Emergency Response Guidelines. Thus the proposed change does not change the likelihood of occurrence of a malfunction of the WHPS to preclude a water hammer event

The Main Control Room control switches (2HS-FW261/262/263/264) for the 2FW035A/B/C/d valves are designed to allow the valves to be held open. Thus the proposed change does not change the likelihood of occurrence of a malfunction of these valves. The associated annunciator circuit and annunciator tile is modified to reflect the new normal position for power operation (above approximately 25%).

The proposed change does not impact the other UFSAR described WHPS features. When the WHPS FW isolation is disabled at power, the water hammer prevention isolation function to preclude injection of cold water into a steam-voided preheater section of the steam generator will continue to be fulfilled by the safety related ESFAS signals to close the feedwater isolation valves on low steam line pressure safety injection or steam generator water level low-low reactor trip coincident with Tave less than 564° F. At low power levels or during startup the WHPS FW isolation features will be enabled.

UFSAR Section 10.4.7.3 will be revised as a result of this activity to describe the capability to disable the WHPS FW isolation control circuits when the Engineered Safety Features Actuation System (ESFAS) Feedwater Isolation feature is active.

Additionally, the proposed change does not revise the operating procedure steps associated with switching feedwater flow from the top feed nozzle to the lower main feed nozzle. As described in the response to FSAR Question 010.51 initial plant startup testing was performed to confirm that no damaging water hammer occurs when FW delivery is transferred from the top SG nozzle to the main feed nozzle. Since the transfer procedure steps are unchanged, there is no change in the frequency of a water hammer when FW delivery is transferred. Therefore the proposed change will not result in more than a minimal increase in the likelihood of a water hammer induced malfunction.

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Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 617266 / DRP 17-023 **Revision Number:** 000 / 0

Title: Provide Capability to Disable Water Hammer Prevention System (WHPS) Feedwater (FW) Isolation Signals During Normal Power Operation & Change WHPS SG Low Pressure/Level Control Relay Operation From Actuate On De-Energized To Actuate On Energized

Based on the above, the proposed activity will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.
**

Disabling the WHPS low SG level and low SG pressure FW isolation auto close signals above approximately 25% power does not introduce the possibility of a change in the consequences of an accident because the FW isolation from the WHPS is not credited with mitigating any accident previously evaluated in the UFSAR. The ESFAS FW isolation signals (Steam line pressure SI or the Reactor trip + Lo Tave) will continue to close the feedwater isolation valves as designed for accident mitigation. Thus the proposed activity will not result in a more than minimal increase in the consequences of an accident previously evaluated in the UFSAR.
**

Disabling the WHPS low SG level and low SG pressure FW isolation auto close signals above approximately 25% power does not introduce the possibility of a change in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR because the FW isolation from the WHPS is not credited to mitigate the consequences of malfunctions of an SSC important to safety. The proposed change does not impact the safety related ESFAS FW isolation circuitry. Thus there is no change to the UFSAR described failure modes and effects for the ESF actuation system. The proposed changes to the WHPS FW isolation auto close signals does not introduce any new failure mode for the system
**

Disabling the WHPS low SG level and low SG pressure FW isolation auto close signals above approximately 25% power does not introduce the possibility of a new accident because the proposed change is not the initiator of any accident and no new failure modes are introduced. When the WHPS FW isolation is disabled at power the automatic water hammer prevention function will be fulfilled by the safety related ESFAS signals for FW isolation. The isolation of FW in events resulting in a reduction of FW temperature without a reactor trip (e.g., turbine trip below P-8 permissive) was never an automatic feature in the WHPS. As per original design, operators will be notified of low FW temperature or low FW flow by Main Control Room annunciators and manual operator action will be taken to correct the condition or to isolate FW flow to the SG preheater section per existing BAR procedures (Ref. BAR 2-15-A11, BAR 2-15-A12, et.al.). This manual operator action to address low FW temperature or flow is not affected by this activity. Administrative controls will be put in place to ensure that the WHPS FW isolation signals are enabled during startup and low power operation. The likelihood of operator error leading to an accident of a different type is not credible because the actions are reflected in plant procedures and operator training programs, the actions can be completed in the required time, the actions are simple and adequate time is available to recover from any credible error. The design bases function to preclude the admission of cold FW into a steam-voided preheater section of the steam generator is not adversely impacted, thus no new accident types are introduced.
**

Disabling the WHPS low SG level and low SG pressure FW isolation auto close signals above approximately 25% power does not introduce the possibility of a malfunction with a different result because when the WHPS FW isolation is disabled at power the water hammer prevention function will be fulfilled by the safety related ESFAS signals. Administrative controls will ensure that the WHPs FW isolation signals are enabled during startup and low power operation
**

This proposed modification of the Unit 2 WHPS low SG level and low SG pressure FW isolation auto close signals does not result in a change that would cause any system parameter to change. WHPS is not credited for mitigation in any accident analysis, thus there is no impact on any UFSAR described fission product barrier limit. Therefore, the proposed activity does not result in a Design Basis Limit for a Fission Product Barrier (DBLFPB) as described in the UFSAR being exceeded or altered
**

This activity modifies the Unit 2 WHPS low SG level and low SG pressure FW isolation circuits to provide the capability to disable the auto close signals when the units are operating above approximately 25% power. The proposed change does not involve a method of evaluation as defined in LS-AA-104-1000. Therefore, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analysis.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): **Byron Unit 2**

Activity/Document Number: **EC 617266 / DRP 17-023** **Revision Number:** **000 / 0**

Title: **Provide Capability to Disable Water Hammer Prevention System (WHPS) Feedwater (FW) Isolation Signals During Normal Power Operation & Change WHPS SG Low Pressure/Level Control Relay Operation From Actuate On De-Energized To Actuate On Energized**

**

Therefore, based on this 50.59 Screening and Evaluation, this activity may be implemented under the governing station procedures without prior NRC approval.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	**6E-17-042**	**Rev.**	**0**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-17-005**	**Rev.**	**0**

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): **Byron/Units 1 and 2**

Activity/Document Number: **EC 400500, TRM Change # 17-004, DRP # 16-030** **Revision Number:** **0/0/0**

Title: **PROCESS TORMIS LAR & SUPPORTING DOCS, TRM 3.7.e "Tornado Design Basis SXCT Fans – Operating", and TRM 3.7.f "Tornado Design Basis SXCT Fans – Shutdown"**

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

The proposed activity involves revision of the licensing and design basis for safety significant components that are not fully protected from tornado-generated missiles. The change involves the following:

1. Analyses that establish missile damage acceptance criteria for the exposed Ultimate Heat Sink (UHS) riser pipes, the exposed 1B/2B Auxiliary Feedwater (AF) diesel engine exhaust, the Main Steam (MS) PORV exhaust tail pipes, and the Miscellaneous Electric Equipment Room (MEER) exhaust openings in the wall between the Auxiliary Building and the Turbine Building (L-line wall). The missile damage analyses for UHS riser pipes and the MS PORV exhaust tail pipes are dispositioned in 50.59 Screening No. 6E-17-047. The missile damage analyses for the 1B/2B Auxiliary Feedwater (AF) diesel engine exhaust and the MEER exhaust openings are dispositioned in 50.59 Evaluation No. 6G-17-004.

2. The Byron tornado missile analysis using the TORMIS methodology. TORMIS uses a Monte Carlo simulation technique to assess, through a Probabilistic Risk Assessment (PRA) methodology, the probability of multiple missile hits causing unacceptable damage to unprotected safety-significant components at a plant. The results from the Byron TORMIS analysis are used to credit the unprotected equipment for post-tornado safe shutdown. A License Amendment Request was submitted and approved to utilize the TORMIS computer code methodology. Thus, this part of the activity is dispositioned in the 50.59 Applicability Review.

3. Revised UHS temperature analysis for safe shutdown from a tornado event. The Essential Service Water Cooling Tower (SXCT) fans and cells that are shown via the TORMIS analysis to survive a tornado strike will be credited for UHS cooling. The revised UHS temperature analysis for safe shutdown from a tornado is dispositioned in 50.59 Evaluation No. 6G-17-004.

4. New TRM 3.7.e, "Tornado Design Basis SXCT Fans – Operating" and TRM 3.7.f, "Tornado Design Basis SXCT Fans – Shutdown" are put in place to ensure that the assumed initial conditions in the post-tornado UHS cooldown analysis are met; i.e. the administrative controls will specify the number of SXCT fans required to be operable based on outside environmental conditions. The new TRMs are dispositioned in 50.59 Screening No. 6E-17-047.

5. Procedure changes to implement the new TRMs and the operator actions assumed to be performed in the supporting analyses. The specific operator action changes are listed in the "Effect of Activity" section of this 50.59 Review Coversheet Form. Procedure changes to implement the new TRMs are dispositioned in 50.59 Screening No. 6E-17-047. The procedure changes for the operator actions are dispositioned in 50.59 Evaluation No. 6G-17-004.

6. UFSAR Changes associated with the TORMIS method and analysis. The TORMIS UFSAR changes are dispositioned in the 50.59 Applicability Review.

7. UFSAR Changes associated with the revised analysis of the UHS for safe shutdown in a tornado event. These UFSAR changes are dispositioned in 50.59 Screening No. 6E-17-047.

8. UFSAR Changes to correct the descriptions associated with the following:

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Units 1 and 2

Activity/Document Number: EC 400500, TRM Change # 17-004, DRP # 16-030 Revision Number: 0/0/0

Title: PROCESS TORMIS LAR & SUPPORTING DOCS, TRM 3.7.e "Tornado Design Basis SXCT Fans – Operating", and TRM 3.7.f "Tornado Design Basis SXCT Fans – Shutdown"

- UFSAR Section 3.5.3 - The description the walls and roofs of structures protecting components from tornado generated missiles is revised to replace the minimum thickness values with a statement that the thicknesses are well above the calculated missile penetration thicknesses.
- UFSAR Section 2.4.11.6 is revised to indicate that the onsite wells are located in reinforced concrete enclosures as opposed to the current description of located in enclosed heated buildings to provide freeze protection.
- UFSAR Sections 3.5.4.2 and 9.5.8.2 are revised to remove the statement: Analysis has established that the stacks can be damaged to the extent that the flow area is reduced to 50% of the original flow area without reducing the diesel power output.

These non-editorial UFSAR changes are dispositioned in 50.59 Screening No. 6E-17-047.

9. Editorial UFSAR Changes associated with the description of tornado winds, tornado missiles, tornado missile protection features, and the UHS. The editorial UFSAR changes are dispositioned in the 50.59 Applicability Review.

10. Procedure changes to remove the compensatory actions put in place associated with for equipment identified as not adequately protected from tornado missiles under Enforcement Guidance Memorandum 15-002. The subject equipment was included in the Byron TORMIS analysis. With the NRC approval of TORMIS and the change to the licensing basis, the EGM 15-002 compensatory actions are no longer required. These procedure changes are dispositioned in the 50.59 Applicability Review.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

In NRC Inspection Report, "Byron Station, Units 1 and 2, Integrated Inspection Report 05000454/2009004; 05000455/2009004," dated November 5, 2009, Byron Station received a non-cited violation, NCV, for failure to protect the Emergency Diesel Generator (EDG) Diesel Oil Storage Tank (DOST) vent lines from tornado-generated missiles. During the extent of condition review to address this violation, additional safety related pipes vulnerable to tornado-generated missiles were identified (i.e., the Steam Generator Power Operated Relief Valve (PORV) tailpipes, the Main Steam Safety Valve (MSSV) tailpipes, and the 1B/2B Auxiliary Feedwater (AF) Diesel exhaust stacks).

Subsequently, in NRC Inspection Report, "Byron Station, Units 1 and 2, NRC Component Design Bases Inspection; Inspection Report 05000454/2015008; 05000455/2015008 and Notice of Violation," dated July 21, 2015, Byron Station received another non-cited violation, for failure to evaluate the adverse effects of changing the SXCT Tornado Analysis as described in the UFSAR, Section 3.5.4, "Analysis of Missiles Generated by a Tornado," Revision 14. This revision changed the UFSAR to assume that two SXCT fans survive a tornado strike. As a result of this violation, the UFSAR analysis of record reverted back to the original licensing basis assumption that multiple tornado missile hits could result in the loss of all SXCT fans. An Operability Evaluation was completed and is currently in place to address the concern that Cold Shutdown cannot be achieved within 72 hours with no SXCT fans available after a postulated tornado event. This Operability Evaluation will be closed out after implementation of this activity.

On May 25, 2016, Byron Station issued Event Notification Report No. 51958, "Discovery of Non-Conforming Conditions During Tornado Hazards Analysis." This Notification Report documents non-conforming conditions in the plant design such that specific Technical Specification equipment on both units was considered to be inadequately protected from tornado missiles. These conditions were addressed in accordance with Enforcement

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): **Byron/Units 1 and 2**

Activity/Document Number: **EC 400500, TRM Change # 17-004, DRP # 16-030** **Revision Number:** **0/0/0**

Title: **PROCESS TORMIS LAR & SUPPORTING DOCS, TRM 3.7.e "Tornado Design Basis SXCT Fans – Operating", and TRM 3.7.f "Tornado Design Basis SXCT Fans – Shutdown"**

Guidance Memorandum 15-002, "Enforcement Discretion for Tornado-Generated Missile Protection Noncompliance," dated June 10, 2015 and DSS-ISG-2016-01, "Clarification of Licensee Actions in Receipt of Enforcement Discretion Per Enforcement Guidance Memorandum EGM 15-002, "Enforcement Discretion for Tornado-Generated Missile Protection Noncompliance."

To resolve the above concerns and close out the Operability Evaluation, Byron Station requested NRC approval to utilize the TORMIS Computer Code methodology for assessing tornado-generated missile protection of the Byron Station SSCs. Unprotected targets needed for safe shutdown after a tornado, including the unprotected UHS components, are included in the TORMIS analysis. The results of the TORMIS analysis specific to the post-tornado survival of the SXCT fans are then used to evaluate SXCT capability for safe shutdown after a tornado event.

NRC approval of the TORMIS methodology for Byron was granted via NRC Safety Evaluation dated 8/10/2017.

This activity implements the change in the licensing and design basis for safety significant components that are not fully protected from tornado-generated missiles. This includes revision of the UHS licensing basis, implementation of new TRMs and procedure changes that support the revised UHS analysis. The UFSAR change package revises the UFSAR to describe application of the TORMIS method, the revised UHS tornado analysis, and makes other UFSAR changes to clarify the description of the design basis for tornado winds, tornado missiles, tornado missile protection features, and the UHS.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

Design bases
The activity revises the design bases associated with the effects of tornado and high wind generated missiles. The design is revised to utilize the TORMIS Computer Code as a methodology for assessing tornado missile protection of selected plant components. The TORMIS analysis utilizes a probabilistic approach to determine the acceptability of the current level of missile protection (or lack of missile protection) for the modeled plant components on each unit. The TORMIS analysis is a supplement to the current licensing/design bases for tornado generated missiles. The original deterministic tornado generated missile design basis still applies to the barriers that provide missile protection to the systems, structures, and components not evaluated using TORMIS.

The original licensing basis assumed that multiple tornado missile hits could result in the loss of all SXCT fans. The application of the TORMIS analysis results revises the design basis for the UHS for post-tornado operation from natural draft cooling to mechanical draft cooling (i.e. from passive to active cooling tower fans).

Safety analyses described in the UFSAR
The UFSAR described safety analysis for the UHS after the loss of SXCT fans due to tornado-generated missiles is revised to credit unprotected SXCT fans that survive a tornado strike based on the results of the TORMIS analysis. The revised UHS analysis considers SXCT fans that may be out of service and a worst case single failure that results in the failure of two SXCT fans. Based on the results of the TORMIS analysis:

- 3 or more SXCT fans survive when one SXCT fan is out of service and two failed SXCT fans are assumed.
- 2 or more SXCT fans survive when two SXCT fans are out of service and two failed SXCT fans are assumed.

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Station/Unit(s): **Byron/Units 1 and 2**

Activity/Document Number: **EC 400500, TRM Change # 17-004, DRP # 16-030** **Revision Number:** **0/0/0**

Title: **PROCESS TORMIS LAR & SUPPORTING DOCS, TRM 3.7.e "Tornado Design Basis SXCT Fans – Operating", and TRM 3.7.f "Tornado Design Basis SXCT Fans – Shutdown"**

The number of SXCT cells that may be initially out of service is dependent on the outside air wet bulb temperature and number of operating units. The revised UHS analysis for a tornado event determined that with the available fans and risers, Cold Shutdown conditions can be reached for both units within 36 hours while maintaining the SX supply water temperature less than or equal to the design maximum temperature of 100° F.

Plant operations
The activity has the following impact on plant operations:

- New operating limits are put in place associated with the number of required Operable SXCT fans based on the outside air wet bulb temperature. New required actions and completion times are established to ensure that the initial conditions assumed in the revised analysis are met.
- The existing time critical operator action to manually initiate the deep well pump(s) within 90 minutes of a LOCA is revised to include a tornado event where SX makeup from the Rock River is not available.
- The existing time critical operator action to isolate SX blowdown within 2 hours of a LOCA is revised to include a tornado event where SX makeup from the Rock River is not available.
- Operator action is required to isolate the AF telltale vent lines is required within 2 hours of a tornado event where SX makeup from the Rock River is not available. This action is currently directed in 1/2BEP ES-0.2, "NATURAL CIRCULATION COOLDOWN", step 21.b, if less than or equal to 4 SXCT fans are running with associated risers open.
- Operator action is required to isolate the SXCT riser leak-off drains within 2 hours of a tornado event that damages SXCT fans. This makes permanent the action put in place under Operability Evaluation 13-007. This action is currently directed in 1/2BEP ES-0.2, "NATURAL CIRCULATION COOLDOWN", step 21.b, if less than or equal to 4 SXCT fans are running with associated risers open and in procedure BOP SX-T2, Step D.4.
- For post-tornado cooldown with damage to the SXCT, AF cooling concurrent with RH cooling will be used until the RCS is cooled below 280° F.
- For a post-tornado event shutdown, operators will use the existing procedural guidance on system temperature limits to control the rate of cooldown. With the postulated minimum number of SXCT fans (fans initially out of service, fans lost to a postulated single failure, and fans damaged by tornado missiles), the cooldown time may approach 36 hours.
- For a tornado event with missile damage to the MEER exhaust, manual operator action is required within 2 hours to restore ventilation cooling to the affected room(s). This action is currently directed in 0BOA ENV-1, Attachment C, Step 2.

Plant operating procedures and operator training programs will be revised to reflect the new or modified operator actions. The following procedures are revised/created:

- 0BOL 7.e and 0BOL 7.f, new procedures associated with the new TRMs.
- 0BOA ENV-1, ADVERSE WEATHER CONDITIONS UNIT 0
- BOP SX-T2, SX TOWER OPERATION GUIDELINES
- 0BOSR 0.1-0, UNIT COMMON ALL MODES/ALL TIMES SHIFTLY AND DAILY OPERATING SURVEILLANCE
- OP-BY-102-106, OPERATOR RESPONSE TIME PROGRAM AT BYRON STATION

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Station/Unit(s): Byron/Units 1 and 2

Activity/Document Number: EC 400500, TRM Change # 17-004, DRP # 16-030 Revision Number: 0/0/0

Title: PROCESS TORMIS LAR & SUPPORTING DOCS, TRM 3.7.e "Tornado Design Basis SXCT Fans – Operating", and TRM 3.7.f "Tornado Design Basis SXCT Fans – Shutdown"

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

In accordance with 10 CFR 50.90 Byron Station requested a license amendment to utilize the TORMIS Computer Code method for assessing tornado-generated missile protection of unprotected plant systems, structures, and components. The LAR stated that the results from the Byron Station TORMIS analysis will be used to credit unprotected equipment for post-tornado safe shutdown. The LAR specifically indicated that the Essential Service Water Cooling Tower (SXCT) fans and cells that survive a tornado strike will be credited for Ultimate Heat Sink (UHS) cooling as opposed to the original licensing basis that assumed all the unprotected SXCT fans are damaged by tornado-generated missiles. The Safety Evaluation dated 8/10/2017 (ML17188A155), approved the requested licensing basis change. Thus, this portion of the proposed activity is controlled under 10 CFR 50.90 and a 50.59 review is not required as noted in the 50.59 Applicability Review.

A number of the proposed UFSAR changes involve reformatting, simplification, and editorial changes. These changes screened out as a Non-Regulatory Change in accordance with the provisions of NEI 98-03, Revision 1 as noted in the applicability review.

Tornado missile damage to the UHS riser pipes that causes less than 50% crimping does not adversely affect performance of the UHS or Essential Service Water (SX) systems. The Essential Service Water Cooling Tower (SXCT) riser pipes are part of the SX system pressure boundary and function to provide a flow path from the underground SX return piping to the SXCT fill distribution piping. Tornado missile damage that causes pipe crimping does not result in loss of the SX system pressure boundary, thus this design function is not adversely affected. Piping flow analysis shows that 50% riser crimping has a negligible impact on system flow rates to components cooled by the SX system and negligible impact to the flow rates to the SXCT fill. The UHS temperature analysis included a sensitivity case to show that the revised flow rates with riser pipe crimping do not adversely impact the cooling function.

Tornado missile damage to the Main Steam (MS) Power Operated Relief Valve (PORV) exhaust tail pipes that causes less than 90% crimping does not adversely affect the relief valve function to relieve steam and maintain the steam generator pressure below the design value. Flow analysis shows that the allowed exhaust pipe crimping has no effect on the valve relieving capacity.

The damage acceptance criteria for tornado missile damage to the 1B/2B AF Diesel engine exhaust does not adversely affect the function of the 1B/2B AF pump, 1B/2B AF cube cooler fan, SX booster pump, the lube oil and the engine fuel oil pumps.

The proposed non-editorial UFSAR changes do not affect the function of any system, structure or component. The UFSAR changes to describe the TORMIS method and analysis reflect the approved licensing basis. The UFSAR changes associated with the revised UHS temperature analysis for a tornado event reflects the revised design basis analysis.

The UFSAR Section 3.5.3 description of the minimum wall thickness for the concrete walls and roofs of structures protecting the safety related systems and components from tornado-generated missiles is revised to remove the specific concrete thickness values. Design analyses show that the various concrete barrier thicknesses are well above the thickness required to prevent spalling and scabbing and the revised description conforms to the requirements of

Section 3.5.3 of Regulatory Guide 1.70, Revision 2. Thus, the proposed UFSAR change to remove the barrier thicknesses has no impact on the UFSAR described missile barrier function.

The proposed UFSAR change to indicate that the onsite wells are located in reinforced concrete enclosures as opposed to located in enclosed heated buildings does not adversely affect the function of the deep well pumps. The UFSAR described design functions are not affected by this UFSAR change because the deep well pump are located below grade in a reinforced concrete enclosure. No heating of the enclosure is required because the well head and discharge piping are located below grade which prevents freezing. The deep well pump enclosures were included as targets in the TORMIS analysis. Based on the results of the TORMIS analysis, the deep well pump enclosures provide an adequate level of protection from tornado missiles and may be credited as an alternate UHS makeup water source after a tornado event.

The proposed UFSAR change to remove the statement that the Emergency Diesel Generator (EDG) exhaust stacks can be damaged to the extent that the flow area is reduced to 50% of the original flow area without reducing the diesel power output does not affect the function of the EDGs. As described in UFSAR Section 3.5.4.2, a rupture disc pressure relief device is installed on each diesel exhaust line. This relief device is located downstream of the silencer and inside the missile protection structure on the roof of the auxiliary building. Upon blockage of the stack, the rupture disc will open prior to backpressure increasing to the point that required diesel power is not available. The emergency diesel generators will therefore remain functional following any postulated tornado missile impact. Since the EDG exhaust system is designed to mitigate blockage of the stack, the description of the impact of a 50% flow area reduction is extraneous information that may be removed from the UFSAR.

The analysis to determine the acceptable amount of tornado missile damage for the UHS riser pipes, 1B/2B AF diesel engine exhaust, and MS PORV exhaust tail pipes does not involve a change to how UFSAR described design functions are performed and controlled. The analyses determine the allowed amount of tornado missile damage to these components. The allowed crimping (reduction in flow area) does not change how the UHS, 1B/2B AF diesel engine, or MS PORVs functions are performed or controlled. Thus, the analyses do not involve a change that adversely affects how a UFSAR described design function are performed or controlled.

The UFSAR changes associated with the revised UHS temperature analysis for a tornado event reflects the revised design basis analysis. Thus, the proposed UFSAR changes do not adversely affect how UFSAR described SSC design functions are performed or controlled.

The change involves flow analysis for the UHS, 1B/2B AF diesel engine exhaust, and MS PORV exhaust. The UFSAR does not describe the method of evaluation for determining flow in the UHS, 1B/2B AF diesel engine exhaust, or MS PORV discharge lines. Thus, the change does not involve an adverse change to an element of an UFSAR described evaluation methodology.

The change involves temperature analysis of the MEER associated with the potential loss of room ventilation. UFSAR Section 3.11.4.2 does not describe the method used to determine MEER room temperatures for a loss of ventilation event. Thus, the change does not involve an adverse change to an element of an UFSAR described evaluation methodology.

The change involves the UFSAR description of the minimum thickness of concrete barriers used for tornado missile protection of safety-related systems and components. As described in UFSAR Section 3.5.3, the depth of penetration into a concrete barrier is calculated using the modified Petry equation and that the concrete barriers are designed such that the missile penetrates no more than two-thirds of the thickness of the barrier thus preventing spalling or

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scabbing. The proposed UFSAR change does not revise, change, or utilize an alternate method for determining the adequacy of the concrete barriers to prevent spalling and scabbing. Thus, the proposed change does not involve an adverse change to an element of a UFSAR described evaluation methodology.

The proposed activity changes the UFSAR described operation of the UHS after a tornado event. The change revises the evaluation to credit forced draft cooling as opposed to operation in strictly natural draft cooling. Use of forced draft cooling of the UHS is currently evaluated for normal and post-LOCA operation. UHS temperature analysis for safe shutdown from a tornado event shows that maximum supply SX water temperature remains within the UFSAR/SER described maximum temperature and supports safe shutdown of both units within the UFSAR described time of 72 hours to reach cold shutdown. Thus, the change does not result in operation of the UHS or the SX system in a manner inconsistent with the descriptions in the UFSAR.

The new tornado design basis operating limits associated with the number of required Operable SXCT fans based on the outside air wet bulb temperature are consistent with current operating limits currently in Technical Specification 3.7.9. Thus, the new TRM limitations do not result in operation or control of the system in a different manner.

Operator actions to maintain UHS inventory and maximize UHS heat removal during a tornado event are consistent with current operating procedures for the UHS during a LOCA event. Existing procedural guidance is used to control the cool down rate and heat input to the UHS. Thus, the proposed changes do not operate the UHS, SX system, or equipment used for safe shutdown in a manner that is different that already described in the UFSAR.

The proposed change evaluates operation of the equipment located in the Miscellaneous Electrical Equipment Rooms after a tornado event for up to 2 hours without room ventilation. UFSAR Section 3.11.4.2 and Table 3.11-2 currently describe MEER room heat-up for a postulated HELB event. The UFSAR describes that room equipment is capable of operating with elevated room temperatures that may occur after a loss of room ventilation. Design analysis has been performed to show that the resultant peak room temperatures will not exceed the EQ temperature limits of the equipment. TRM 3.7.d, Area Temperature Monitoring, allows the room temperature in the MEER and Battery Rooms to exceed the normal maximum design temperature for a limited period of time. Thus, the proposed change does not result in operation of a SSC in a manner that is inconsistent with the UFSAR description or outside the reference bounds of the design.

The proposed UFSAR changes, revised design analysis, new TRM operating limitations, and associated procedure changes do not require a change to Technical Specifications, because the changes do not alter or modify any existing Technical Specification.

The proposed changes associated with the analysis of tornado missile damage and the revised UHS temperature analysis for safe shutdown from a tornado event do not introduce the possibility of a change in the frequency of an accident because the revised analyses do not involve the initiator of any accidents. The proposed changes in operator actions mitigate the consequences of postulated tornado missile damage. The proposed operator actions are not the initiator of any accidents and are simple such that operator errors are judged to not be credible. The proposed changes do not impact the normal or accident function of the 1B/2B AF pump diesel engine, equipment in the MEER or Battery Rooms, or the UHS. The proposed activity is associated with the tornado missile impact on the 1B/2B AF diesel exhaust pipe and cover, postulated missile damage to the MEER and Battery Room exhaust ducts, and the UHS temperature analysis for safe shutdown from a tornado event. UFSAR changes will be made under DRP No. 16-030 to describe the revised analyses. Thus, the proposed activity does not increase the frequency of an accident previously evaluated in the UFSAR.

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Design analysis DO-EDS-1 determined the amount of crimping that would be acceptable for a tornado missile hit on the 1B/2B AF diesel engine exhaust pipes and exhaust cover plates. The allowed amount of crimping is used in design analysis ARA-002116, Tornado Missile TORMIS Analysis for Byron Generating Station. The allowed amount of crimping may result in higher engine exhaust back pressure. The higher engine exhaust back pressure with crimping was evaluated and could result in a slight reduction in the engine power output (~0.65% reduction). There is approximately 12.7% margin between the full load on the 1B/2B AF diesel engine (AF pump full load HP, AF room cubicle cooler fan HP, AF lube oil and fuel pumps, and the SX booster pump HP) and the continuous break horsepower rating for the engine. Additionally, for safe shutdown from a tornado, the 1B/2B AF pump load on the engine will be reduced after a short time as SG level recovers and 1B/2B AF flow is throttled to match the heat removal rate via steaming. The potential 0.65 % reduction in 1B/2B AF diesel engine output due to missile damage on the exhaust pipe or cover plate has a negligible effect on the 1B/2B AF engine and the 1B/2B AF pump. Thus, the allowed amount of tornado missile hit crimping on the 1B/2B AF exhaust pipe or cover plate does not result in a more than minimal increase the likelihood of a malfunction of the 1B/2B AF diesel.

For the MEER HVAC openings, the TORMIS model used "pipe penetration" as failure as opposed to a missile hit on the opening. Using this approach, TORMIS determined that the equipment within the MEER rooms could be credited for safe shutdown from a tornado. The dampers in the wall penetration could be potentially damaged by tornado missiles such that exhaust path is blocked and the affected MEER and Battery Room could loss ventilation cooling. Design Analysis BYR16-048 evaluates room temperatures during this loss of ventilation event. The analysis assumes that operator action will be taken within 2 hours to mitigate the loss of ventilation. Design Analysis EQC-BB-008 previously evaluated the safety related equipment in the MEERs following a HELB event and determined that the components can withstand a 2-hour exposure of 165° F. BYR16-048 determined that the MEER Division 1 maximum temperature at 2 hours would be 140° F and the MEER Division 2 maximum temperature at 2 hours would be 159° F. Since the calculated maximum MEER room temperatures remains below the evaluated temperature of the equipment in the room, the potential loss of ventilation does not adversely impact the function of the equipment important to safety located in the MEER rooms.

The post-tornado temperature excursion in the Battery Rooms was also evaluated. The calculated maximum peak Battery Room temperature was 119°F. The post-tornado increase in room temperature is within the limits of TRM.7.d, which allows the Battery Room temperatures up to 138° F for less than 8 hours. The loss of room ventilation for 2 hours would not result in an unacceptable build-up of hydrogen with the battery rooms as indicated in design analysis NED-H-MSD-017, Revision 4. The analysis concludes upon a loss of ventilation it takes 17.4 hours to reach 2% hydrogen concentration in the most limiting battery charge mode. Thus, the potential loss of ventilation does not adversely impact the function of the equipment important to safety located in the Battery Rooms.

Operator action is credited within two hours to mitigate the MEER and Battery Room heat-up if tornado missile damage results in loss room ventilation. The new action (including the required completion time) is added to plant operating procedures and operator training programs. The required manual action involves simple activities to open doors (to provide an exhaust path) and restart supply fans which can be completed within the required time. Environmental conditions in the plant once the tornado/storm has passed would not challenge performance of this operator action. No credible errors are associated with performing this manual action. Thus, the new operator action would not result in a more than minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety.

The revised UHS temperature analysis for safe shutdown from a tornado event credits new and existing operator actions to minimize the SX supply water temperature. The new or modified operator actions (including the required completion time) supporting the UHS temperature analysis are added to plant operating procedures and operator training programs. Based on validation by Operations, the required actions can be completed in the time required and the resources available. Environmental conditions in the plant and outdoors once the tornado/storm has passed would not challenge performance of the new operator actions. The proposed actions to start the deep well pump(s), isolate SX blowdown, isolate the AF telltale vent lines, and isolate the SXCT riser leak-off drains are simple such that operator errors and recovery actions are judged to not be credible. Operators already control the cool down rate during normal plant shutdowns to maintain RH, CC, and SX temperatures within operating limits. Thus, the new and modified operator actions for the UHS would not result in a more than minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety.

The proposed changes associated with the analysis of tornado missile damage and the revised UHS temperature analysis for safe shutdown from a tornado event do not introduce the possibility of a change in consequence of an accident because the change involves the evaluation of safe shutdown after a tornado event. Based on the results of the approved TORMIS analysis, SXCT fans and cells that survive a tornado strike are credited for UHS cooling. The post-tornado UHS temperature analysis considers a limiting failure and shows that with the available UHS and risers cold shutdown conditions can be reached for both units well before 72 hours while maintaining the SX supply water temperature less than or equal to the maximum design temperature of 100° F.

The proposed changes in operator actions mitigate the consequences of postulated tornado missile damage. The proposed operator actions are not the initiator of any accidents and are simple such that operator errors are judged to not be credible.

Design analysis of the equipment important to safety located within the MEER and Battery Rooms and the AF Diesel Engine will continue to support safe shutdown with the missile damage acceptance criteria utilized in the TORMIS analysis. Thus, the consequences of a tornado event are unchanged. The proposed changes do not affect the operation of any SSC for other UFSAR accidents. Thus, there is no increase in the consequences of an accident previously evaluated in the UFSAR.

The revised UHS temperature analysis for safe shutdown from a tornado event credits SXCT fans and cooling tower cells that the TORMIS analysis indicates will survive a tornado event. The existing licensing basis assumed that all SXCT fans were damaged by tornado missiles and that the SXCT operated strictly in a natural draft mode. When operated strictly in the natural draft mode, malfunctions of the SXCT fans did not need to be considered because the fans were already assumed to be lost to tornado missiles. The revised analysis considers a bounding electrical bus failure that results in the loss of power to two SXCT fans and a deep well pump. The analysis shows that with the available SXCT fans and risers, cold shutdown conditions can be reached for both units well before 72 hours while maintaining the SX supply water temperature less than or equal to the maximum design temperature of 100° F. Design analysis of the equipment important to safety located within the MEER and Battery Rooms and the 1B/2B AF Diesel Engine will continue to support safe shutdown with the missile damage acceptance criteria utilized in the TORMIS analysis. The proposed operator actions to mitigate postulated tornado missile damage are simple such that operator errors are judged to not be credible. The potential for MEER ventilation system operation with only the supply fans operating in the outside air mode post-tornado has minimal impact on the control room pressure boundary and the environment in the control room. Thus, the consequences of a tornado event are unchanged and there is no increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The proposed change associated with the evaluation of the impact of tornado missiles does not introduce the possibility of a new accident and no new failure modes are introduced. The acceptance criteria used in TORMIS for missile damage to the 1B/2B AF diesel engine exhaust has a negligible effect on the 1B/2B AF engine and the 1B/2B

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AF pump. The room heat up due the potential loss of MEER ventilation that could result from the impact of tornado missiles does not result in room temperatures that exceed the design limits for equipment important to safety within the rooms. The 2 hour loss of ventilation in the Battery Rooms will not result in exceeding the hydrogen concentration limits within the rooms or the temperature design limits for the batteries.

The revised UHS temperature analysis considers a bounding electrical bus failure that results in the loss of power to two SXCT fans and a deep well pump. UFSAR Table 9.2-16, already describes this bounding failure mode. Thus, the change does not introduce an accident different from any previously evaluated in the UFSAR.

The proposed activity adds new procedure instructions and operator actions in the event tornado missile damage affects the MEER exhaust dampers and/or SXCT cooling tower fans. The new procedure instructions and operator actions are included in the operating procedures and are simple tasks. Operator errors are judged to not be credible. Thus, the change does not introduce an accident different from any previously evaluated in the UFSAR.

The allowed amount of tornado missile damage to the 1B/2B AF Diesel engine exhaust cover plate does not introduce the possibility for a malfunction of an SSC with a different result because the activity does not introduce a failure result. The acceptance criteria used in TORMIS for missile damage to the 1B/2B AF diesel engine exhaust has a negligible effect on the 1B/2B AF engine and the 1B/2B AF pump.

The potential loss of MEER ventilation due to tornado missile damage to the dampers in the L-line wall does introduce the possibility of a malfunction of an SSC with a different result because the activity does not introduce a failure mode that is not bounded by those described in the MEER ventilation system failure analysis in UFSAR Table 9.4-14. The identified failure mode for this activity is a loss of the air flow path due to a missile impact on the exhaust opening. The failure modes and effects analysis for the system in UFSAR Table 9.4-14 includes loss of airflow or flowpath (fan motor continues to operate). A comparison of these indicates that the results of the failure modes resulting from tornado missile damage to the dampers in the L-line wall is consistent with those presented in the UFSAR.

The proposed change does not introduce the possibility of a malfunction of an SSC with a different result because the activity does not introduce a failure mode that is not bounded by those described in the UHS single failure analysis in UFSAR Table 9.2-16. The revised UHS temperature analysis considers a bounding electrical bus failure that results in the loss of power to two SXCT fans and a deep well pump which bounds the failure modes listed in UFSAR Table 9.2-16. The single failures listed in UFSAR Table 9.2-16 are evaluated for the revised UHS tornado event response in EC 400500. The review determined that the failure modes are bounded by the assumed limiting failure and those presented in the UFSAR.

The revised UHS temperature analysis for safe shutdown from a tornado event shows that with missile damage and the limiting postulated electrical bus failure, cold shutdown conditions can be reached for both units well before 72 hours while maintaining the SX supply water temperature less than or equal to the maximum design temperature of 100° F. The analysis also indicates that the units may be placed on shutdown cooling as early as 7 hours as long as the rate of cooldown is controlled. The proposed new manual operator actions outside of the control room are limited and can be completed in the time required considering the aggregate effects. Thus, the UHS supports the applicable Branch Technical Position RSB 5-1 design requirements for safe shutdown from a design basis tornado event.

The changes to system parameters were evaluated and the changes do not adversely impact the function of equipment required for safe shutdown from tornado event. Therefore, the proposed activity does not result in a DBLFPB as described in the UFSAR being exceeded or altered.

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The revised UHS temperature analysis for safe shutdown from a tornado event credits SXCT fans and cells that the TORMIS analysis indicates will be available. This change in the licensing basis to allow credit for some unprotected SXCT fans and cells based on the results of the TORMIS analysis was approved via a NRC Safety Evaluation for Byron Amendment No. 199, dated 8/10/2017. The revised analysis evaluates basin temperature by performing a minute by minute mass and energy balance on the SXCT basin. The specific method used to calculate the maximum basin temperatures in the original analysis was not described the in the UFSAR (FSAR question response) or SER. The revised analysis uses SXCT performance curves generated using the method described in UFSAR Section 9.2.5.3.1.1.2. The specific method used to determine tower performance in the original analysis was not described in the original FSAR question response or SER. The method used for the revised analysis has been previously been reviewed and accepted by the NRC to evaluate tower performance for the Byron UHS design basis event (LOCA coincident with a LOOP on one unit and concurrent orderly shutdown of the other unit). The revised UHS temperature analysis assumes that for failed fans that have open riser valves, the SXCT cells operating in the natural draft mode provide 10 percent of the heat removal of a cell with the fan running in high speed. Use of 10% cooling for SXCT operation in the natural draft mode is currently described in UFSAR Section 9.2.5.3.1.1.3 and was reviewed and accepted by the NRC in the Safety Evaluation for Byron Station Amendment No. 173. The revised UHS temperature analysis utilizes inputs and assumptions consistent with the original UFSAR described analysis. Thus, the change in to the UHS temperature analysis for safe shutdown from a tornado event does not result in a departure from a method of evaluation described in the UFSAR.

Based on the result of the Applicability Review, 50.59 Screening, and 50.59 Evaluation the activity may be implemented per plant procedures.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☒	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	6E-17-047	**Rev.**	0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-17-006	**Rev.**	0

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): 6G-17-007

Activity/Document Number: EC 621593 **Revision Number:** 0

Title: Lost Parts Eval For Reactor Vessel Head Lift Rig Retaining Ring

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

The activity will evaluate the consequences of a C-shaped retaining ring on the Reactor Coolant system and connected systems if the lost part is not recovered. The lost part has dimensions of 1.775 inches by 1.391 inches with a thickness of 0.150 inches; the ring material is consistent with series 300 stainless steel.

Reason for Activity:
(Discuss why the proposed activity is being performed.)

During an inspection of the Unit 2 Reactor Vessel lifting rig, a retaining ring (1 of 3) was missing during an as-found check; the retaining ring could not be found. It is unknown how long the part has been not installed. Since the lost part could migrate to the RCS via the refueling cavity, ER-AA-2006 requires an evaluation to determine the consequences of the lost part on plant equipment.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

There is no effect on the operation of any system. The location of the retaining ring is not known so it is assumed for this evaluation that it could reach the RCS. The potential for damage to plant components is considered less than minimal based on the retaining rings small size.

Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

ECs 621593 and 621600 evaluated the effects of the retaining ring on RCS, ECCS and fuel components. The lost part will not increase the frequency of occurrence of a stuck control rod, RCP locked rotor, RCP seal failure, RCS decreased flow, or cause a steam generator tube rupture. Since the likelihood of an accident is not increased and the components needed to mitigate the consequences to an accident are not more than minimally impacted, the proposed activity does not result in a more than minimal increase in the consequences of an accident previously evaluated in the UFSAR.

If the retaining ring is transported to the RCS, the potential to damage various applicable components within the RCS and interconnecting systems was evaluated. It was concluded the foreign material does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. The introduction of the retaining ring to the reactor may cause fretting of the fuel cladding, however UFSAR section 4.2 states dose limits given in 10CFR100 and 10CFR50.67 will not be exceeded due to a small number of rod failures. The likelihood of multiple fuel rod failures is unlikely. Therefore, the fuel, vessel, RCS, and connected systems and components would not malfunction in a manner different than has been previously evaluated in the UFSAR. Thus, the consequences of a malfunction are not changed.

The retaining ring is small and would have minimal impact. The lost part does not provide a means to initiate an accident of a different type than any previously evaluated in the UFSAR. The components have been evaluated in ECs 621593 and 621600, and it has been determined that they would continue to perform their design functions and any operational disturbance would be within the plant's design and operational basis. Thus, the lost part would not result in an SSC malfunctioning with a different result than what was previously evaluated in the UFSAR.

The three fission product barriers (fuel clad, RCS, containment) were considered. A fuel failure in itself will not exceed or alter a design basis limit for a fission product barrier (DBLFPB) a fuel failure in itself will not exceed or alter a design basis limit for a fission product barrier (DBLFPB) because per UFSAR section 4.2, dose limits given in 10CFR100 and

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): 6G-17-007

Activity/Document Number: EC 621593 **Revision Number:** 0

Title: Lost Parts Eval For Reactor Vessel Head Lift Rig Retaining Ring

10CFR50.67 will not be exceeded due to a rod failure. Therefore, the lost part do not result in a DBLFPB as described in the UFSAR being exceeded or altered.

The lost part evaluation does not involve any evaluation methodologies used to establish design bases or safety analyses. It is concluded the lost part will not cause more than a minimal impact on plant components and systems.

Based on this evaluation, the proposed activity can be performed without prior NRC permission per the applicable governing procedures.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

- ☐ **Applicability Review**
- ☐ **50.59 Screening** **50.59 Screening No.** ________ **Rev.** ________
- ☒ **50.59 Evaluation** **50.59 Evaluation No.** 6G-17-007 **Rev.** 0

See LS-AA-104, Section 5, Documentation, for record retention requirements for this and all other 50.59 forms associated with the Activity.