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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, 2020 through November 30, 2020 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 Ms. Lisa Zurawski, Regulatory Assurance Manager, at (815) 406-2800.

Respectfully,

H Welt

Harris Welt
Plant Manager
Byron Generating Station

HW/LZ/tv

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-18-001	1	Upgrade Existing AVR with Digital ABB Unitrol Model
2	6G-18-002	0	Westinghouse Ovation Digital Upgrade for 7300 NSSS Cabinets 1(2)PA05J, 1(2)PA06J, 1(2)PA07J, 1(2)PA08J/Westinghouse Ovation Digital Upgrade for 7300 BOP Cabinets 1(2)PA20JA and 1(2)PA20JB/Westinghouse Ovation Digital Upgrade for TDFWP Cabinets 1(2)FW36J and 1(2)FW37J
3	6G-18-003	1	Configuration Control Processing of MSIV Rooms HELB Calculations/Short-Term Pressurization Subcompartment Analysis of Main Steam Tunnel and Main Steam Isolation Valve Rooms
4	6G-19-001	0	Reclassify ASME III FP Piping/Valves/Components in Seismically Qualified Areas
5	6G-19-003	0	FWRV Solenoid SPV elimination
6	6G-19-004	1	Install/Remove Temp Control Loops for 7300 NSSS and BOP Cabinets TCCP Temporary
7	6G-19-005	0	Secondary Pump Trip Unit 1/2
8	6G-19-006	0	Replacement of System Aux. Transformer (SAT) 242-2 with ABB Transformer Unit 2
9	6G-20-001	0	Lost Parts Evaluation for 2B Charging Pump Mechanical Seal Debris

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 618133 (U2) and EC 618132 (U1) / DRP 17-042 (U2) and DRP 17-043 (U1)
Revision Number: 002 and 000 / NA for DRPs

Title: Upgrade Existing AVR with Digital ABB Unitrol Model

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 Westinghouse / Cutler-Hammer solid-state main generator automatic voltage regulator (AVR) with a new Asea Brown Boveri (ABB) digital AVR, Model Unitrol 6000 Medium, which performs the same critical functions in the overall excitation system as the existing AVR. In place of the single-channel design of the existing AVR, the replacement AVR uses a two-channel design to improve reliability. The AVR will continue to receive power from the permanent magnet generator (PMG). In addition, two independent 480 Vac power feeds are introduced to support the dual-channel design to enhance reliability. The PMG will supply one channel; one of the 480 Vac power feeds will supply the other channel, while the second 480 Vac will provide a backup source of power that can be manually aligned to supply the first channel (normally supplied by the PMG). Since the new AVR also includes a power system stabilizer (PSS), the existing PSS panel is removed. The new design modifies the main control room controls and provides an excitation control terminal (ECT) touchscreen panel at the AVR for local control and monitoring. Existing power sources are re-utilized where necessary but new 480 Vac power sources and an additional 125 Vdc feed are required for this modification.

DRP 17-042 (U2) and DRP 17-043 (U1) are issued in conjunction with this activity to update the main exciter description in UFSAR Section 10.2.2.2 to indicate that the generator exciter will be controlled by a digital AVR equipped with two redundant channels.

Note, 50.59 Evaluation No. 6G-18-001 Revision 1 was issued to reflect update of the AVR Electromagnetic Compatibility (EMC) assessment.

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

The existing AVR is obsolete and is no longer supported by the manufacturer. The new AVR is a state-of-the-art system offering improved reliability.

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

Instead of separate base adjuster and voltage adjuster controls, there will be one control to adjust voltage. The AVR will automatically line up the voltage of the automatic and manual control modes. Instead of being switched on at a specified power level, the power system stabilizer function will be actuated automatically. Control board indications, switches, and alarms will be modified to reflect the new operational requirements of the replacement AVR system. A new excitation controls terminal (ECT) will be installed at the AVR cabinet – this terminal can only be operated under strict administrative controls.

There is no impact on 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.)

Failures in the AVR system could result in a turbine trip and challenges to the offsite power system. The replacement AVR system includes digital hardware and software, significant changes to the human-machine interface, and the automation of functions previously performed manually. Since there is a potential to adversely affect UFSAR-described design functions, these aspects of the proposed activity were "screened in" for further assessment under the 50.59 Evaluation process.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 618133 (U2) and EC 618132 (U1) / DRP 17-042 (U2) and DRP 17-043 (U1)
Revision Number: 002 and 000 / NA for DRPs

Title: Upgrade Existing AVR with Digital ABB Unitrol Model

Like the existing AVR, failures in the replacement AVR could result in a turbine trip and challenges to the offsite power system. The failure modes and effects of the replacement AVR are bounded by those of the existing AVR. The replacement AVR is a state-of-the art system widely used in the industry and is provided by a vendor with considerable experience. The replacement AVR incorporates a dual-channel design in place of the single-channel design of the existing AVR, and additional sources of power have been provided by the station to support the dual-channel design. The operator interface with the AVR system has been simplified, and existing station practices ensure that the operators are familiar with the replacement system and with the required interface with the system.

The improvements in reliability provide assurance that there is no more than a minimal increase in the frequency of occurrence of an accident or in the likelihood of a malfunction previously evaluated in the UFSAR. Previous analyses of events which could result from an AVR malfunction remain bounding; therefore, the radiological consequences of accidents or malfunctions are not affected. The AVR system does not interface with any Ovation-based control systems, so there is no potential for a common-cause failure affecting multiple plant systems that could create the possibility for an accident or malfunction not previously evaluated.

The proposed activity does not affect a design basis limit for a fission product barrier. Supporting analyses/evaluations for this activity have been performed in a manner consistent with standard industry practices and consistent with the evaluation requirements / methodologies described in the UFSAR. This activity does not involve a test or experiment not described in the UFSAR. The proposed activity does not affect the Technical Specifications or the Facility Operating License.

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-18-018	**Rev.**	0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-18-001	**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 & 2

Activity/Document Number: EC 617681 (U1) and EC 617685 (U2) / EC 617682 (U1) and EC 617686 (U2) / EC 617670 (U1) and EC 617671 (U2) / DRP 18-006 (U1) and DRP 18-007 (U2) **Revision Number:** 000 and 000 / 000 and 000 / 000 and 003 / NA for DRPs

Title: Westinghouse Ovation Digital Upgrade for 7300 NSSS Cabinets 1(2)PA05J, 1(2)PA06J, 1(2)PA07J, 1(2)PA08J / Westinghouse Ovation Digital Upgrade for 7300 BOP Cabinets 1(2)PA20JA and 1(2)PA20JB / Westinghouse Ovation Digital Upgrade for TDFWP Cabinets 1(2) FW36J and 1(2)FW37J

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 upgrades nuclear steam supply system (NSSS), balance of plant (BOP), and turbine-driven feedwater pump (TDFWP) control systems by modifying individual control systems and incorporating those individual control systems into a plant-wide distributed control system (DCS). The activity includes various control system changes to incorporate improvements and lessons learned.

The activity encompasses the following engineering changes:

- EC 617681 and EC 617685 replace the 7300-series equipment in NSSS cabinets PA05J, PA06J, PA07J, and PA08J. These ECs also include changes to the main control board (MCB) to support the modifications to these NSSS cabinets.
- EC 617682 and EC 617686 replace the 7300-series equipment in BOP cabinets PA20JA and PA20JB. These ECs also include changes to the MCB to support the modifications to these BOP cabinets.
- EC 617670 and EC 617671 eliminate the 7300-series equipment in TDFWP cabinets PA36J and PA37J and install new local cabinets FW36J and FW37J. These ECs also include both changes to the main control board (MCB) to support the modifications to these cabinets and changes to the TDFWP turbine control and protection system hardware.

The proposed activity involves the instrument loops for the following systems, with a wide range in the level of complexity:

1. Distributed Control System (DCS)
2. Reactor Coolant System (RC/RY)
3. Nuclear Instrumentation System (NR)
4. Chemical and Volume Control System (CV)
5. Residual Heat Removal (RH)
6. Safety Injection (SI)
7. Containment Spray (CS)
8. Component Cooling (CC)
9. Auxiliary Feedwater (AF)
10. Main Steam (MS)
11. Main Feedwater (FW)
12. Heater Drain (HD)
13. Condensate / Condensate Booster (CD/CB)
14. Primary Water (PW)
15. Main Generator (HY/WS)
16. Main Turbine (MT)
17. Essential Service Water (SX)
18. Circulating Water (CW)
19. Liquid Radwaste (RF/WX)

The specific changes to these systems are described in the 50.59 screening.

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

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 617681 (U1) and EC 617685 (U2) / EC 617682 (U1) and EC 617686 (U2) / EC 617670 (U1) and EC 617671 (U2) / DRP 18-006 (U1) and DRP 18-007 (U2) **Revision Number:** 000 and 000 / 000 and 000 / 000 and 003 / NA for DRPs

Title: Westinghouse Ovation Digital Upgrade for 7300 NSSS Cabinets 1(2)PA05J, 1(2)PA06J, 1(2)PA07J, 1(2)PA08J / Westinghouse Ovation Digital Upgrade for 7300 BOP Cabinets 1(2)PA20JA and 1(2)PA20JB / Westinghouse Ovation Digital Upgrade for TDFWP Cabinets 1(2) FW36J and 1(2)FW37J

The activity is part of an overall phased project to upgrade key process control systems – integrating them into a DCS and eliminating the existing Westinghouse 7300-series process control systems – to address equipment reliability and obsolescence issues. The activity also includes various control system changes to incorporate improvements and lessons learned, based on operating experience with the existing control systems.

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

The activity involves numerous changes to the operator interface with the affected systems. The specific changes are described in the 50.59 screening.

The upgrade and changes to the control systems do not affect the design bases. The changes were reviewed with respect to the safety analyses described in the UFSAR and it was concluded that no changes to the safety analyses are required and the existing safety analyses remain bounding.

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 installation of a substantial amount of digital hardware and software, as well as significant changes to the human-machine interface. In accordance with the guidance provided in NEI 01-01, these aspects of the proposed activity were "screened in" for further evaluation under the 50.59 process. In addition, certain functional changes to the affected control systems were screened in:

- elimination of: auctioneered-high signals (Tavg, auctioneered-high nuclear power, and auctioneered-high delta-T) as inputs to major plant control systems, and the feedwater flow input to the heater drain tank level control circuit
- installation of automatic features in place of manual actions involving: RCS cooldown when Tavg is below 550°F, control of pressurizer pressure below 1700 psig, and several aspects of the main feedwater system

The rigorous process used in developing the digital hardware and software and the integrated testing of the major control systems using a plant-specific model were credited with ensuring the modified systems would perform as required. The involvement of the plant operations staff in the development of the human-machine interface, the work with Idaho National Laboratory in reviewing operator interaction with the new equipment at their simulation laboratory, the simulator testing at both Westinghouse and Exelon, and the Braidwood operator training process were credited with ensuring a successful operator interface with the new system. As a result, 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 or more than a minimal increase in the likelihood of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Since the existing safety analyses for accidents that could be initiated by failures in the control systems remain bounding, no radiological consequences beyond the current consequences for such events would occur. Therefore, the proposed activity does not result in more than a minimal increase in the consequences of accidents or malfunctions and does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. The proposed activity does not involve a method of evaluation described in the UFSAR.

A review of the failure modes and effects analysis for the activity indicates that that the upgrade of the control systems has eliminated many single point vulnerabilities and those few single failures which could lead to a significant power transient or other serious effect on the plant are present in the existing control systems. A software hazards analysis also evaluated controller

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 617681 (U1) and EC 617685 (U2) / EC 617682 (U1) and EC 617686 (U2) / EC 617670 (U1) and EC 617671 (U2) / DRP 18-006 (U1) and DRP 18-007 (U2) **Revision Number:** 000 and 000 / 000 and 000 / 000 and 003 / NA for DRPs

Title: Westinghouse Ovation Digital Upgrade for 7300 NSSS Cabinets 1(2)PA05J, 1(2)PA06J, 1(2)PA07J, 1(2)PA08J / Westinghouse Ovation Digital Upgrade for 7300 BOP Cabinets 1(2)PA20JA and 1(2)PA20JB / Westinghouse Ovation Digital Upgrade for TDFWP Cabinets 1(2) FW36J and 1(2)FW37J

malfunctions with respect to existing UFSAR Chapter 15 events to determine whether transient and accident analyses might be affected. The system-level failure analysis did not identify malfunctions (software hazards) that affect the transient and accident analysis in Chapter 15 of the UFSAR. The likelihood of a common cause failure is considered sufficiently low based on the design attributes of the system (e.g., preventive, limiting, and likelihood-reduction measures), the quality of the design processes employed, and operating experience in similar applications. Therefore, the evaluations of single failures and of the potential for common cause failures provide an adequate basis for concluding that the proposed activity does not create a possibility for an accident of a different type or a malfunction with a different result than any previously evaluated in the UFSAR.

Design basis limits for fission product barriers are not affected by the proposed activity. There is no adverse change to an element of a UFSAR-described evaluation methodology, or use of an alternative methodology, that is used in establishing the design bases or used in the safety analyses.

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-18-042	**Rev.**	1
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-18-002	**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

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

Activity/Document Number: EC 404484 / DRP #17-056 **Revision Number:** 000 / 000

Title: CONFIGURATION CONTROL PROCESSING OF MSIV ROOMS HELB CALCULATIONS / Short-Term Pressurization Subcompartment Analysis of Main Steam Tunnel and Main Steam Isolation Valve Rooms

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 change in the analysis of the short-term pressures in the Main Steam Tunnel (MST) and Main Steam Isolation Valve (MSIV) Room subcompartments following a high energy line break (HELB) as documented in EC 404484. It includes UFSAR Change (DRP) #17-056 which updates the UFSAR sections that provide the details of the updated analysis, such as locations and numbers of design basis breaks postulated, and the peak differential pressure results for the MST and MSIV Room subcompartments. The UFSAR update also includes clarification of the areas of system piping where no pipe breaks are postulated to occur, and to remove excessive detail regarding the methodology, assumptions, analytical model, and other details of the analysis.

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

UFSAR Section 3.6 describes the protection against dynamic effects associated with the postulated break of piping. One effect is described as structural loading due to pressurization resulting from a pipe break. Calculation 3C8-0282-001, Rev. 003 calculates the short-term peak and transient pressures in the Main Steam Tunnel (MST) and Main Steam Isolation Valve (MSIV) Room subcompartments following a high energy line break (HELB). Results from this calculation are discussed in UFSAR Section 3.6 and UFSAR Attachments A3.6 and C3.6. Design errors have been identified in this short-term pressurization subcompartment analysis. These errors are documented in several AR's (e.g., IR 4085634 and 4085649 for Byron). In order to resolve the identified errors, calculation 3C8-0282-001 has been updated and determines new short-term peak and transient pressures in the Main Steam Tunnel (MST) and Main Steam Isolation Valve (MSIV) Room subcompartments following a high energy line break (HELB).

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 determine bounding mass & energy (M&E) releases for each MST and MSIV Room subcompartments and to determine the peak and transient short-term pressures and update the UFSAR with the new results. There is no impact on plant operations or response to any accidents. The change to the UFSAR provides the updated safety analysis as described in UFSAR Section 3.6.

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.)

An Applicability Review was performed and determined a portion of the activity involves a change to the UFSAR that removes excessive detail as discussed in NEI 98-03. The information removed is design information from UFSAR Attachment C3.6 that is not important to the description of the facility or of the 10CFR50.2 design bases or safety analyses of the facility.

Screening Conclusions

The new thermal-hydraulic analysis for evaluating the short-term pressurization event in the MST and MSIV Room subcompartments contained in calculation 3C8-0282-001, Rev. 004 and implemented via EC 404484, involves the use of an alternative methodology from what was previously used in establishing the effect of the design bases HELBs in these areas. This is evaluated in the 50.59 Evaluation in Question 8.

All other aspects of the proposed activity, including resolution of the errors in the short-term pressurization subcompartment analysis, do not alter the UFSAR described function of any SSCs.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 404484 / DRP #17-056 Revision Number: 000 / 000

Title: CONFIGURATION CONTROL PROCESSING OF MSIV ROOMS HELB CALCULATIONS / Short-Term Pressurization Subcompartment Analysis of Main Steam Tunnel and Main Steam Isolation Valve Rooms

There is no adverse effect on the way that the UFSAR described design function is performed or controlled. There is no testing involved with this change.

This change does not require a change to the Technical Specifications or Operating License.

Evaluation Conclusions

The attached evaluation addresses the effect involving the use of an alternative methodology from what was previously used in establishing the effect of design bases HELBs in the MST and MSIV Room subcompartments. However, 50.59 Evaluation 6G-18-003, determined that the use of GOTHIC to evaluate the effects of HELBs in the MST and MSIV Room subcompartments, rather than RELAP, is not departure from a method described in the UFSAR because it is appropriate for the intended application and the method has been approved by the NRC.

Based on the above, and as detailed in the attached 50.59 screening and evaluation, this activity may proceed per normal plant processes and procedures without requesting prior approval from the NRC.

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-18-043	Rev.	0
☒	50.59 Evaluation	50.59 Evaluation No.	6G-18-003	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 Station/Unit 0,1,2**

Activity/Document Number: **EC 626662/DRP 18-004** **Revision Number:** **000/NA**

Title: **Reclassify ASME III FP Piping / Valves / Components in Seismically Qualified Areas**

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.)

Engineering Change (EC) 626662 is a Design Change Document Change Request (DCR) that incorporates the supporting documentation for the reclassification of Fire Protection (FP) piping and components in the Auxiliary Building, Fuel Handling Building and Containment from Safety Category I, Quality Group C (ASME Section III, Class 3) to Safety Category II, Quality Group D (Non-Safety Related ANSI B31.1) with the exception of the containment piping from check valves 1(2)FP345 up to and including relief valves 1(2)FP360 and selected other supports which will remain classified as Safety Category I, Quality Group C. All the affected FP piping and components will continue to be classified as Seismic Category I. The classification of the containment isolation portion of the FP System which is Safety Category I, Quality Group B (ASME Section III, Class 2) is not affected by this DCR.

The Safety Classification of the FP System SSCs impacted by this DCR, as documented in PassPort, will be Augmented Quality (AQ) in accordance with CC-AA-304 and NO-AA-10, Section A.2.4 to meet 10 CFR 50 Appendix A, General Design Criterion (GDC) 3. This classification is consistent with the remainder of the FP System in non-seismically qualified areas. In addition, the affected SSCs have the augmented requirement of being qualified as Seismic Category I.

The documentation changes incorporated under this DCR include drawing changes, a new calculation, UFSAR DRP 18-004, PassPort equipment/component data changes, and piping specification changes. Fire Protection Program changes, such as the revision to the Fire Protection Report, are evaluated separately under LS-AA-128. Inservice Inspection Plan changes are evaluated separately under its associated program.

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

A portion of the FP System piping at the Byron Generating Station was conservatively classified as Safety Category I, Quality Group C during plant construction. This classification maintained consistency between the Seismic Category I requirements for the FP System as delineated in the Fire Protection Report and the Safety Category classification requirements included in UFSAR Section 3.2.1.1. This classification, however, imposes unnecessary ASME code requirements and special treatment under 10 CFR 50 Appendix B during preventive & corrective maintenance, design change control, procurement, work control, testing, quality inspection and documentation for components that do not perform a function important to safety . The changes incorporated under EC 626662 relax process burden by controlling the FP System commensurate with the Safety Category II, Quality Group D functions it performs. This permits utilizing plant resources to focus on safety significant SSCs.

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

The changes incorporated under this DCR do not change the function of the FP System SSCs or the functions supported by the FP System. The FP System continues to provide a Seismic Category I standpipe system to supply hose stations within the Auxiliary Building, Fuel Handling Building and Containment which can be supplied from the Essential Service Water System (SX), if necessary, during a safe shutdown earthquake event. The Seismic Category I standpipe system remains capable of providing a backup source of makeup water to the spent fuel pool. There is no change to the alternate cooling water supply provided by the standpipe system for the Chemical and Volume Control System (CV) Pumps. The portion of the FP System that provides containment isolation and penetration over-pressure protection is not changed by this DCR.

The analytical methods of demonstrating FP SSC structural qualification and ability to remain functional during design basis events are unchanged. The Seismic Category I classification of these SSCs is maintained. Therefore, the structural requirements for a Safety Category I remain applicable to the affected portion of the FP System that is being reclassified to Safety Category II. As such, no undesirable structural interaction is created with interfacing Safety Category I SSCs such as the supply from the SX System and the FP System containment penetration.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): **Byron Station/Unit 0,1,2**

Activity/Document Number: **EC 626662/DRP 18-004** **Revision Number:** **000/NA**

Title: **Reclassify ASME III FP Piping / Valves / Components in Seismically Qualified Areas**

No physical or functional changes are implemented under EC 626662. Repair and replacement activities following the implementation of this DCR will be governed by ANSI B31.1 and the augmented quality requirements associated with the Fire Protection Program and the maintenance of the Seismic Category I classification of the affected SSCs. These requirements establish the necessary process controls and treatment for the system with respect to maintaining pressure boundary integrity, structural integrity and functionality following a safe shutdown earthquake event.

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 following is a summary of the conclusions in Screening 6E-19-005:

Although the screening and justification included in EC 626662 concluded that the reclassification of the seismically qualified portions of the FP System to Safety Category II, Quality Group D was appropriate, the change was considered adverse. The UFSAR specifically references the Safety Category I classification of the FP System in the discussion of its design functions. The changes ultimately relax process requirements and special treatment under the station's 10 CFR 50 Appendix B program for future repair and replacement activities that could potentially reduce FP System reliability which in turn could impact other safety related SSCs.

The document changes incorporated under EC 626662 do not change procedures that impact the operation of the FP System as described in the UFSAR.

The UFSAR described evaluation methodology associated with the piping, components, and supports for the FP System in seismically qualified areas is unaffected by the changes implemented under EC 626662.

This activity does not represent a test or experiment. There are no physical, analytical, or operational changes being implemented for the FP System that would be inconsistent with the analyses or descriptions in the UFSAR.

The FP System containment isolation function associated with Technical Specifications is not impacted by EC 626662. In addition, the containment penetration over-pressure protection provided by relief valves 1(2)FP360 are unaffected by this DCR. No Technical Specification or Facility Operating Changes are necessary to implement the changes under EC 626662.

Based on the adverse change in component classification to Safety Category II, Quality Group D, it was concluded that the activity required a 10 CFR 50.59 Evaluation.

The following is a summary of the conclusions in Safety Evaluation 6G-19-001:

As justified in EC 626662 and Screening 6E-19-005, the changes incorporated under EC 626662 do not represent a departure from design, fabrication, construction, testing, and performance standards specified in the General Design Criteria and other regulatory requirements for the appropriate Safety Category II, Quality Group D classification of the affected portion of the FP System. EC 626662 does not introduce the possibility of a change in the frequency of an accident because the seismically qualified portion of the FP System and associated SSCs is not an initiator of any accident and no new failure modes are introduced.

The affected portion of the FP System will meet the design requirements for material and construction practices commensurate with its non-safety related design function. Safety analyses are unaffected and the proposed changes will not degrade the system performance or reliability. Therefore, the proposed activity will not increase the likelihood of a malfunction.

Since FP System integrity is maintained and containment penetration and SX System integrity is unaffected by the proposed change, the change will not result in an increase in the consequences of previously analyzed accidents.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): **Byron Station/Unit 0,1,2**

Activity/Document Number: **EC 626662/DRP 18-004** **Revision Number:** **000/NA**

Title: **Reclassify ASME III FP Piping / Valves / Components in Seismically Qualified Areas**

There are no physical changes introduced by the proposed activity. The consequences of malfunctions previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The FP System is not credited to perform any accident mitigation functions as documented in the UFSAR.

The augmented controls maintained for the affected SSCs, Seismic Category I and as required by GDC 3, ensure no new failure modes are created that could impact or reduce the availability, operability, or effectiveness of equipment important to safety and introduce an accident different from any previously evaluated in the UFSAR.

Since no new failure modes are introduced by the proposed activity, there is no potential for a different result associated with a malfunction of the seismically qualified portions of the FP System affected.

The reclassification of the seismically qualified portion of the FP System under EC 626662 does not result in a change that would cause any system parameter to change. The changes implemented under the proposed activity do not impact a fission product barrier. Therefore, the proposed does not result in design basis limit for a fission product barrier (DBLFPB) as described in the UFSAR being exceeded or altered.

The UFSAR described evaluation methodology associated with the piping, components, and supports for the FP System in seismically qualified areas is unaffected by the changes implemented under EC 626662. The changes 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.

Based on the above, 10 CFR 50.59 Evaluation concluded that the activity can 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.**	**6E-19-005**	**Rev.**	**000**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-19-001**	**Rev.**	**000**

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 & 2

Activity/Document Number: EC# 624969, EC# 627386 **Revision Number: 0, 0**

Title: FWRV Solenoid SPV Elimination

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 proposed activity modifies the Feedwater Isolation Circuit for the Feedwater Regulating Valves (_FW510, _FW520, _FW530 and _FW540) from a de-energize-to-actuate scheme to an energize-to-actuate scheme. As a result, the FWRVs will no longer close on loss of power to the solenoids. A solenoid failure would be isolated to one train and the opposite train would still be able to provide the desired isolation function. Wiring modifications are made to revise the control and test circuitry to an energize to actuate circuit.

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

The reason for the proposed activity is to reduce the inadvertent and unwanted spurious actuation of FWRV closure from a single equipment failure in the Control Power Source for the actuation circuit. The present design of the Feedwater Isolation Scheme being de-energize-to-actuate design, a single equipment failure in a redundant control power supply can result in unwanted FWRV closure, resulting in a plant trip.

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

The proposed activity only affects the FWRVs on loss of power. A solenoid failure and a loss of power in the FWRV isolation circuit would be isolated to one train and the opposite redundant train would still be able to provide the desired isolation function. The design chosen for energize-to-actuate is like the Feedwater Isolation Circuit used on Byron's FW009 Feedwater isolation valves. Changing the solenoids to energize-to-actuate does not interfere with the ability of the FWRVs to close on receipt of an ESF actuation signal.

UFSAR 7.1.2.6 will be updated to reflect the proposed activity that changes the "de-energize-to-actuate" logic to "energize to actuate" logic for loss of power (DRP# 18-010 and DRP# 18-011).

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 various accident analyses were reviewed for potential impact for probability of occurrence. The only accident that revising the feedwater isolation circuit for the FWRV could potentially impact is loss of normal feedwater flow (Chapter 15.2.7) due to inadvertent closure of the FWRV. The revised control circuit decreases potential for these failure mechanisms causing an inadvertent actuation and the loss of normal feedwater flow event analyzed in UFSAR accident analysis. In the existing design, the solenoid coils are continuously energized. This presents two potential failure mechanisms that are not present in the revised design.

FWRV as part of FW isolation valves are credited in UFSAR Table 15.0-7 to mitigate the effects of various accidents postulated in UFSAR Chapter 15. The proposed activity only affects the FWRVs on loss of power. Changing the solenoids to energize-to-actuate does not interfere with the ability of the FWRVs to close on receipt of an ESF actuation signal as credited in UFSAR Chapter 15. UFSAR Chapter 15.2.7 discusses pump failures, valve malfunctions, or loss of offsite AC power as the initiator for a loss of normal feedwater event, which reduces the capability of secondary system to remove the heat generated in the reactor core. The proposed activity is intended to reduce the occurrence of inadvertent FWRV closure and loss of normal feedwater event by redundancy provided in control power and solenoid. The fail-safe design on loss of power, for the affected FWRV isolation circuit does not affect the existing FMEA for ESFAS as evaluated in the UFSAR. The proposed activity does not

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

Activity/Document Number: EC# 624969, EC# 627386 **Revision Number: 0, 0**

Title: FWRV Solenoid SPV Elimination

introduce any new failure modes previously not evaluated for the ESFAS, but rather the "energize to actuate" principle increases the reliability of the FWRV isolation circuit to actuate when required.

Feedwater isolation is credited in the accident analysis UFSAR Chapter 15.1 Increase in heat removed by the secondary system and Chapter 15.2, Decrease in heat removed by the secondary system. The proposed activity affects the FWRVs on loss of power. Changing the solenoids to energize-to-actuate does not interfere with the ability of the FWRVs to close on receipt of an ESF actuation signal as credited. The significant characteristics, parameters and assumptions for feedwater isolation in mitigating these accidents are actuation setpoints, timing of the actuation function, and reliability of the isolation function. The proposed design does not in any way impact the reactor protection system established limits for the various actuation signals nor does it impact the timing of an isolation signal to the solenoid valves. The revised design is as reliable in providing a required isolation as the existing design, as it remains electrically supplied by reliable Safety Related DC power. Thus, for those analyses crediting feedwater isolation for accident mitigation, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

The proposed design scheme eliminates the loss of normal feedwater flow event (and thereby any resulting radiological consequences) related to loss of power to the FWRV solenoid coils. Changing the solenoids to energize-to-actuate does not interfere with the ability of the FWRVs to close on receipt of an ESF actuation signal.

This proposed activity revises the design scheme for closure of the FWRV on a feedwater isolation signal to prevent inadvertent closure of FRW. It does not change any of the conditions, bases or events that require feedwater isolation. The proposed change does not affect the normal or safety functions of the FWRV. Inadvertent closure of FWRV is already an analyzed condition in the UFSAR, loss of normal feedwater flow.

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 UFSAR. The proposed activity only affects the FWRV on loss of power. Changing the solenoids to energize-to-actuate does not interfere with the ability of the FWRVs to close on receipt of an ESF actuation signal. Since no new failure modes are identified per the Failure Modes Design Evaluation documented in EC#624969, no new malfunction initiator is created by this proposed activity.

The overall function of the FWRV is unaffected by the proposed change, so there is no impact on any actuation setpoint(s), actuation timing, or actuation. Therefore, this activity does not result in a change that would cause any system parameter to change. Therefore, this proposed activity does not result in a DBLFPB as described in the UFSAR being exceeded or altered.

This proposed activity to modify the Feedwater Isolation Circuit for the Feedwater Regulating Valves (_FW510, _FW520, _FW530 and _FW540) from a de-energize-to-actuate scheme to an energize-to-actuate scheme does not involve a method of evaluation as defined in the UFSAR.

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-19-003**	**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 Units 1 & 2

Activity/Document Number: EC 617683 (U1) and EC 617688 (U2) **Revision Number:** 001/000

Title: Install/Remove Temp Control Loops for 7300 NSSS and BOP Cabinets TCCP Temporary

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 temporary configuration change (TCC) that will be performed in support of separate activities which are upgrading multiple nuclear steam supply system (NSSS) and balance of plant (BOP) control systems and transferring them from the existing 7300-series system to an Ovation-based distributed control system (DCS) under EC 617681, EC 617685, EC 617682, and EC 617686.

The TCC involves the transfer of instrument loops which are to be upgraded but which are also required and/or desired to be available during the transition of the plant into the refueling outage. During the refueling outage, the permanent changes to the NSSS and BOP control systems will be made.

The proposed activity involves the installation of:

- a temporary control panel (TCP) in the auxiliary electrical equipment room (AEER) that will house the necessary controllers and input/output (I/O) modules
- temporary cables from the existing NSSS (PA05J, PA06J, PA07J, PA08J) and BOP (PA20JA, PA20JB) cabinets to the TCP for the affected instrument loops
- two temporary workstations in the control room and one temporary workstation in the radwaste control room
- connections of the temporary workstations and TCP to the Ovation network

This configuration will connect the affected instrument loops to the Ovation-based DCS and allow for their operation via the temporary workstations. This will ensure that key process variables can be monitored and controlled during the transition to and from the EC 617683 and EC 617688 implementing refueling outages. The TCC allows work required for the permanent upgrade of the NSSS and BOP control systems to the Ovation-based DCS to be initiated prior to commencement of the implementing refueling outages. The TCP will be removed once it is no longer required, and the affected control systems will then be a part of the permanently upgraded NSSS and BOP control systems.

The proposed activity involves the following systems:

1. Distributed Control System (DCS)
2. Reactor Coolant System (RC/RY)
3. Chemical and Volume Control System (CV)
4. Residual Heat Removal (RH)
5. Safety Injection (SI)
6. Component Cooling (CC)
7. Condensate (CD)
8. Primary Water (PW)
9. Essential Service Water (SX)
10. Circulating Water (CW)
11. Liquid Radwaste (WX)

The proposed activity does not involve the major NSSS or BOP control systems (i.e., control of Tavg, pressurizer pressure / level, steam dumps, steam generator level, feedwater pump speed, or heater drain level). The proposed activity does not involve the reactor protection or engineered safety features actuation systems. Most of the affected instrument loops are used to monitor the status of equipment and provide indication and alarm functions only. The affected instrument loops with the potential to impact plant processes – such as boric acid flow, charging flow, or VCT level – are not transferred until the reactor is shut down and the RCS is borated to the cold shutdown condition. Some of the affected instrument loops do involve UFSAR-described design functions, as discussed in the 50.59 screening.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 617683 (U1) and EC 617688 (U2) **Revision Number:** 001/000

Title: Install/Remove Temp Control Loops for 7300 NSSS and BOP Cabinets TCCP Temporary

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

The activity is part of an overall phased project to upgrade key process control systems – integrating them into a DCS and eliminating the existing Westinghouse 7300-series process control systems – to address equipment reliability and obsolescence issues. This TCC will ensure that key process variables can be monitored and controlled during the transition to and from the EC implementing refueling outages. The TCC allows work required for the permanent upgrade of the NSSS and BOP control systems to the Ovation-based DCS to be initiated prior to commencement of the implementing refueling outages.

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

The TCC will install a temporary control panel in the AEER, two temporary workstations in the main control room, and a temporary workstation in the radwaste control room. Most of the affected instrument loops are used to monitor the status of equipment and provide indication and alarm functions only. These instrument loops will be transferred to the TCP 1-4 weeks before the refueling outage. The affected instrument loops with the potential to impact plant processes – such as boric acid flow, charging flow, or volume control tank level – are not transferred until the reactor is shut down and the RCS is borated to the cold shutdown condition.

The TCC will ensure that key process variables can be monitored and controlled during the transition to and from the refueling outage via the temporary workstations. The transfer of the affected instrument loops to the TCP will result in the loss of certain main control board indication, control, or alarm (annunciator) functions. In some cases, the alarm or control functions are maintained via the Ovation-based system. If an existing alarm function triggers an Ovation-based alarm, that alarm will also actuate the "Ovation System Trouble" annunciator. In addition, reactor operators will periodically monitor the affected instrument loops for proper behavior and will respond to Ovation alarms per approved station procedures.

Operator time-critical actions involving the affected instrument loops were evaluated to ensure that operator actions credited in the design and licensing basis can be accomplished within the required times.

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 installation of a substantial amount of digital hardware and software, as well as significant changes to the human-machine interface. In accordance with the guidance provided in NEI 01-01, these aspects of the proposed activity were "screened in" for further evaluation under the 50.59 process for Question 1 regarding a change to an SSC that adversely affects an UFSAR-described design function and Question 2 regarding a change to a procedure that adversely affects how UFSAR-described SSC design functions are performed or controlled. The 50.59 Screening concluded in the response to Questions 3, 4 and 5 that the proposed activity: does not involve an adverse change to an element of a UFSAR-described evaluation methodology nor use an alternate evaluation methodology used in establishing the design bases or used in the safety analyses, does not involve a test or experiment described in the UFSAR where an SSC is utilized or controlled in a manner that is outside of the reference bounds of the design for that SSC, and does not require a change to the Technical Specifications or Facility Operating License.

The rigorous process used in developing the digital hardware and software were credited in the 50.59 Evaluation with ensuring the modified systems would perform as required. The involvement of the plant operations staff in the development of the human-machine interface and the Byron operator training process were credited with ensuring a successful operator interface with the new system. As a result, the proposed activity will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR or more than a minimal increase in the likelihood of a malfunction of an SSC important to safety previously evaluated in the UFSAR. Since the affected instrument loops with the potential to impact plant

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 617683 (U1) and EC 617688 (U2) **Revision Number:** 001/000

Title: Install/Remove Temp Control Loops for 7300 NSSS and BOP Cabinets TCCP Temporary

processes – such as boric acid and primary water flow, charging flow, or volume control tank level – are not transferred until the reactor is shut down and the RCS is borated to the cold shutdown condition, the accidents with the potential to be impacted are limited to boron dilution events, and the possibility for an accident of a different type is not created. Since certain instrument loops used for control board indications and alarms will be transferred to the temporary controls during power operation, operator time-critical actions involving the affected instrument loops were evaluated to ensure that operator actions credited in the design and licensing basis can be accomplished within the required times, such that there is no increase in the radiological consequences of accidents previously evaluated in the UFSAR. No radiological consequences would occur, and no design basis limit for a fission product barrier as described in the UFSAR is exceeded or altered. Since administrative controls to isolate the RCS from potential sources of unborated water will be implemented in Modes 3, 4, and 5 to ensure that an inadvertent dilution cannot occur, the possibility for a malfunction with a different result is not created. The proposed activity does not involve a method of evaluation described in the UFSAR.

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-19-017	**Rev.**	0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-19-004	**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 & Braidwood/Units 1&2

Activity/Document Number: 1/2BOA SEC-1, 1/2BwOA SEC-1 **Revision Number:** 114/115/112/110

Title: Secondary Pump Trip Unit 1/2

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 revises abnormal operating procedures 1/2BOA SEC-1 and 1/2BwOA SEC-1 to include actions to operate the Auxiliary Feedwater (AF) system during unit startup in Mode 1(reactor power <15% and prior to synchronization of the main generator to the grid) and Mode 2 to address a loss of the normal feedwater (FW) flow without requiring a reactor trip. Current procedural direction requires a manual reactor trip when the normal FW system is lost in Modes 1 and 2. The procedure changes being implemented under this Activity will direct Operations to start the AF system and stabilize the unit at a power level in Mode 2 within the capability of the AF pumps following a loss of the normal FW with reactor power initially less than 15% during a unit startup and prior to synchronization of the main generator to the grid. Operation will continue in Mode 2 until the normal FW system can be restored or the unit will be shutdown if required by operational or secondary chemistry conditions. Continued operation without hydrazine injection with the AF system feeding steam generators is limited to 8 hours.

In accordance with current abnormal procedures, the reactor will be manually tripped if the normal FW flow is completely lost in Mode 1 at power levels greater than 15%. Procedural direction already exists in 1/2BOA SEC-1 and 1/2BwOA SEC-1 to operate the AF system to address a loss of the normal FW flow in Mode 3.

As documented in the supporting 50.59 Evaluation No. 6G-19-005/BRW-E-2019-32, technical justification to support the operation of the Auxiliary Feedwater (AF) system in the event that normal feedwater is lost during a unit startup prior to synchronizing the main generator to the grid is provided in EC 628034 for Byron and EC 627959 for Braidwood.

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

The purpose of the procedure changes is to manually start the AF pumps and not immediately trip the reactor following a loss of the normal feedwater during unit startup in Mode 1(at reactor <15% and prior to synchronization of the main generator to the grid) and Mode 2. During startup, the normal FW flow is initially from only one FW pump placed in service. In some circumstances, the availability of the other FW pumps may be limited due to continuation of maintenance following scheduled or forced outages. This condition reduces the normal redundancy provided by the FW system during the unit startup and increases the possibility of a loss of normal FW to occur.

The procedure changes will allow operators to stabilize steam generator levels with the AF system and eliminate need to trip the reactor, which in turn eliminates the plant transient associated with a reactor trip.

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

The procedure changes implemented under this Activity will allow Operators to stabilize steam generator levels using the AF system following a loss of the normal FW flow in Mode 1(reactor power <15% and prior to synchronization of the main generator to the grid) and Mode 2 and restore a normal FW flow source without placing an operational transient on the plant resulting from a reactor trip. Since current procedural direction and automatic design features exist to start the AF pumps following the loss of the normal FW flow and reactor trip, there is no change in the demands on the AF system and the AF system would be operated consistent with the current procedures and design basis.

The loss of the normal FW flow is an accident analyzed in Chapter 15 of the UFSAR. However, the accident is analyzed for the bounding case with reactor power initially at 100% of rated power. The analysis considers a reactor trip occurring on low steam generator level and an automatic start of the AF pumps. The UFSAR does not specifically analyze a loss of the normal FW flow at low power/start-up since this condition would be bounded by the 100% event. The use of the AF system to pre-emptively respond to stabilize steam generator levels is not considered in the UFSAR analyzed event since the AF system does not have adequate flow capacity to support plant operation at full power.

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

Activity/Document Number: 1/2BOA SEC-1, 1/2BwOA SEC-1 **Revision Number:** 114/115/112/110

Title: Secondary Pump Trip Unit 1/2

The design flow capability of the AF system is based on a feedline break downstream of the feedwater isolation valves to prevent an over-pressurization of the Reactor Coolant System (RCS). During the response to the upset condition of a loss of normal feedwater at low power, it may be necessary to stabilize steam generator levels by manually adjusting AF system flow to less than that assumed in the UFSAR Chapter 15 analysis for a feedline break event. Similarly, it may be necessary to stabilize steam generator levels by manually adjusting AF system flow to greater than that assumed in the Steam Generator Tube Rupture Margin-to-Overfill analysis in UFSAR Chapter 15. The loss of feedwater is a Category II event as discussed in the UFSAR Chapter 15. Additional concurrent design basis events are not postulated.

For postulated piping failures, the AF system was not evaluated/designed as a high-energy system based on guidance established in BTP MEB 3-1. Specifically, since the AF system is not operated during normal plant startup and since the fraction of time that the AF system operates within the pressure-temperature conditions specified for high-energy fluid system is less than 2 percent of the time that the system operates, the AF system was classified as a moderate-energy system. The guidance in the BTP MEB 3-1 includes amplification that systems such as the reactor decay heat removal system qualify as moderate-energy fluid systems based on limited operation at high-energy conditions; however, systems such as auxiliary feedwater systems operated during normal PWR reactor startup, hot standby, or shutdown would still qualify as high-energy fluid systems.

The procedure changes implemented under this Activity do not change the requirement to use the AF system in response to a loss of normal FW. The change is that the reactor is not tripped before starting the system for the upset condition resulting from a loss of normal FW in Mode 1(reactor power <15% and prior to synchronization of the main generator to the grid) and Mode 2. The AF system would still be used as before to control steam generator levels following the loss of normal FW event. Therefore, the procedural changes would not significantly increase the time that the AF system is operated at a high-energy condition. The procedure changes do not allow use of the AF system in lieu of the normal FW source for plant startup. As documented in EC 628034 and EC 627959, the AF system is capable of supporting a steady-state reactor power level of 3% with a single AF pump in operation and 6.8% reactor power with two AF pumps in operation. Given the limited capacity of the AF system, continuing with the startup is not practical and procedure steps will require the normal FW to be restored before power ascension is continued. The changes to the procedures will direct actions following a loss of the normal FW system to stabilize steam generator levels until the normal FW system can be restored. This application is consistent with the use of the AF system following any loss of normal FW system transients, where the AF system is used to maintain decay heat removal with plant in hot standby or shutting down. Therefore, the change does not require re-evaluation of the AF system for postulated high energy pipe breaks since the AF system is operated during upset conditions only.

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 following is a summary of the conclusions in Evaluation 6G-19-005/BRW-E-2019-32:

The changes to the procedures used in response to loss of the normal FW system in Mode 1(reactor power <15% and prior to synchronization of the main generator to the grid) and Mode 2 do not change the frequency of occurrence of a loss of the normal FW system. The changes are intended to eliminate the plant transient associated with a manual reactor trip. Since the plant is maintained at normal conditions for the startup and the reactor trip eliminated, there are no additional challenges placed on the plant that could lead to accidents. In response to a loss of the normal FW system in Mode 1(reactor power <15% and prior to synchronization of the main generator to the grid) and Mode 2, the AF system will be operated in a manner consistent with its design capabilities. The operator actions necessary to control reactor power in the response to the event are not complex and do not increase the likelihood of errors which could increase the frequency of reactivity events. Therefore, the change is not considered to change the frequency of occurrence for any accidents evaluated in the UFSAR.

Since current procedures start or verify the start of the AF system following a loss of the normal FW system and reactor trip, there is no change in the demands on the AF system and the AF system would be operated consistent with the current procedures and design basis. Therefore, there is no change in the likelihood of a failure with AF system components, including the likelihood of pipe breaks. The proposed procedure changes do not degrade the performance of a safety system

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

Activity/Document Number: 1/2BOA SEC-1, 1/2BwOA SEC-1 **Revision Number:** 114/115/112/110

Title: Secondary Pump Trip Unit 1/2

assumed to function in the safety analyses below the level of performance assumed in the safety analyses. The changes do not degrade the performance of the AF system such that it cannot perform its safety functions at the reactor power levels associated with the change. The subsequent restoration of normal FW does not increase the likelihood of a FW system malfunction as analyzed in the UFSAR. The operator burden associated with the proposed procedure changes do not increase the likelihood of an SSC malfunction previously evaluated in the UFSAR. A limiting duration for operating in Mode 2 with the AF system in operation is provided to prevent the degradation of steam generator tube integrity due to injecting water that is not treated with hydrazine.

The loss of normal FW flow is an accident analyzed in Chapter 15 of the UFSAR. The accident is analyzed for the bounding case with reactor power initially at 100% of rated power. The analysis considers a reactor trip occurring on low steam generator level and an automatic start of the AF system. The UFSAR does not specifically analyze a loss of the normal FW system at low power/start-up since this condition would be bounded by the 100% event. The use of the AF system to pre-emptively respond to stabilize steam generator levels is not considered in the UFSAR analyzed event since the AF system does not have adequate flow capacity to support operation at full power. In the 100% power event, the delay in establishing AF system flow was limiting for decay heat removal and for maintaining acceptable plant conditions. The procedure changes would manually start the AF system prior to low steam generator levels occurring and would maintain normal steam generator levels. The plant remains operating under normal parameters so there are no consequences for the loss of the normal FW system in Mode 1(reactor power <15% and prior to synchronization of the main generator to the grid) and Mode 2. If the AF system is not able to maintain steam generator levels, the operator will proactively initiate a manual reactor trip in response to the degrading level trend prior to reaching the Lo-2 reactor trip setpoint. In the absence of any operator action, the reactor will trip automatically on Lo-2 steam generator level and automatically start the AF system. However, the AF system would already be running and there would be no delays in establishing AF flow for heat removal. Since the currently analyzed loss of normal FW accident bounds the condition considered in this Evaluation, the dose consequences are not impacted by the proposed Activity.

The design flow capability of the AF system is based on a feedline break downstream of the feedwater isolation valves to prevent an over-pressurization of the RCS. During the response to the upset condition of a loss of normal feedwater, it may be necessary to stabilize steam generator levels by manually adjusting AF system flow to less than assumed in the UFSAR Chapter 15 analysis for a feedline break event. Similarly, it may be necessary to stabilize steam generator levels by manually adjusting AF system flow to greater than that assumed in the Steam Generator Tube Rupture Margin-to-Overfill analysis in UFSAR Chapter 15. However, these design basis events are not postulated concurrent with the upset condition of a loss of normal feedwater during unit startup and the dose consequences of these events are not affected.

Therefore, the changes in the procedures to manually start the AF system prior to tripping the reactor on a loss of the normal FW system during unit startup in Mode 1(reactor power <15% and prior to synchronization of the main generator to the grid) and Mode 2 will not change the consequence of the loss of normal FW, feedline break, or steam generator tube rupture accidents as evaluated in the UFSAR.

The procedure changes implemented under this Activity do not change the requirement to use the AF system in response to a loss of normal FW, only that the reactor is not tripped before starting the system for the upset condition resulting from a loss of the normal FW system. Therefore, the AF system would still be used as before to control steam generator levels following the loss of normal FW event. Therefore, the procedural changes would not significantly increase the time that the AF system is operated at a high-energy condition. The procedure changes do not allow use of the AF system in lieu of the normal FW source for plant startup. Given the limited capacity of the AF system, continuing with the startup is not practical and procedure steps will require the normal FW system to be restored before power ascension is continued and will not change use of the AF system for normal startup. The changes to the procedures will direct actions following a loss of the normal FW system to stabilize steam generator levels until the normal FW system can be restored. This application is consistent with the use of the AF system following any loss of the normal FW system transients, where the AF system is used to maintain decay heat removal with plant in hot standby or shutting down. The operation of the AF system in response to the loss of the normal FW system in Mode 1(reactor power <15% and prior to synchronization of the main generator to the grid) and Mode 2 does not introduce any new system failure modes and the dose consequences of any accident that the AF system is designed to mitigate is unchanged. Therefore, the changes implemented under the proposed Activity will not result in more than a minimal increase in the dose consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR

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

Activity/Document Number: 1/2BOA SEC-1, 1/2BwOA SEC-1 **Revision Number:** 114/115/112/110

Title: Secondary Pump Trip Unit 1/2

The proposed procedure changes impact the abnormal operating procedures used for a loss of the normal FW system events. The loss of the normal FW system is already an accident considered in the UFSAR. The accident is analyzed for the bounding case at an initial power level of 100% of rated power and does not specifically analyze a loss of the normal FW system at low power/start-up since this condition would be bounded by the 100% event. The change in the procedures to manually start the AF system prior to tripping the reactor on loss of the normal FW system in Mode 1(reactor power <15% and prior to synchronization of the main generator to the grid) and Mode 2 is in response to the loss of the normal FW system to stabilize and maintain steam generator levels and does not cause the loss of the normal FW system.

The procedural strategy for a loss of normal feedwater utilizes the AF system to maintain steam generator levels and a heat sink for the reactor. At low reactor power levels, in particular with the reactor critical and power less than the point of adding heat (POAH), the injection of colder water into the steam generators from the Condensate Storage Tank has the potential to result in an excessive RCS cooldown and reactivity transient. To address this, the procedures establish manual reactor trip criteria based on startup rate and RCS temperature limitations consistent with existing guidance. In addition, prior to the start of the first AF pump, the operators will be cautioned to limit the injection flow rate to ensure that an excessive RCS cooldown does not occur. The concern for excessive cooldown with reactor power greater than the POAH is minimal based on the AF system injection flowrates being less than or equal to the normal FW system flowrate prior to its loss and the differential temperatures between the AF and FW systems. At very low power levels, the flow and temperature difference is insignificant. At higher reactor power levels (<15%) and for extended unloaded turbine operation, the temperature difference increases, but the AF system injection flow rates are significantly less than the pre-event normal FW system flow rates such that there is no RCS cooldown. In addition, due to the initial loss of normal FW, RCS temperature increases until either the condenser steam dump system responds or AF system injection turns temperature. Therefore, the proposed procedural strategy, the operational limitations established for a manual reactor trip, and the plant's inherent response results in a minimal impact on RCS temperature and core reactivity when the AF system is used to recover from a loss of normal FW during startup with reactor power less than 15% and prior to synchronizing the main generator tó the grid including Mode 2 operation above and below the POAH.

Therefore, the procedure changes do not create a new accident from those already considered in the UFSAR.

For postulated piping failures, the AF system was not classified as a high-energy system based on guidance established in BTP MEB 3-1. Specifically, since the fraction of time that the AF system operates within the pressure-temperature conditions specified for high-energy fluid system is less than 2 percent of the time that the system operates, the AF system was evaluated as a moderate-energy system. The guidance in the BTP MEB 3-1 includes amplification that systems such as the reactor decay heat removal system qualify as moderate-energy fluid systems based on limited operation at high-energy conditions; however, systems such as auxiliary feedwater systems operated during PWR reactor startup, hot standby, or shutdown would still qualify as high-energy fluid systems. The proposed procedure changes direct actions following a loss of the normal FW system in Mode 1(reactor power <15% reactor power) and Mode 2, an upset condition, to stabilize steam generator levels until the normal FW system can be restored in Mode 2. The procedure changes do not allow use of the AF system in lieu of the normal FW source for normal plant startup. Given the limited capacity of the AF system, continuing the startup is not practical and procedure steps will require the normal FW system be established before power ascension is continued beyond Mode 2 and will not change use of the AF system for normal start-up. This application is consistent with the use of the AF system following any loss of the normal FW systems, where the AF system is used to maintain decay heat removal with the plant in hot standby and shutdown and does not create any new failure modes for the system. Therefore, the changes do not require re-evaluation of the AF system for postulated high energy pipe breaks since the procedure changes are not altering the design basis status of the AF system as a moderate energy system and the changes will not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR. Furthermore, no credible operator errors could result in reactivity addition rates that exceed those evaluated in Section 15.4 of the UFSAR. Therefore, the proposed procedure changes do not produce a different result or change the consequences of reactivity anomalies at low reactor power levels as described in the UFSAR.

The changes to the procedures will modify the response to the actions taken in response to loss of the normal FW system during startup in Mode 1(reactor power <15% and prior to synchronization of the main generator to the grid) and Mode 2. These changes do not alter any fission product barriers. In addition, the changes to the procedures did not require a change to any of the analyses used to establish the design basis limits for fission product barriers. 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.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: 1/2BOA SEC-1, 1/2BwOA SEC-1 **Revision Number:** 114/115/112/110

Title: Secondary Pump Trip Unit 1/2

No design basis analyses were revised for the procedure changes. Postulated line breaks in the AF system were originally evaluated as a moderate-energy system since the operation of AF system at high-energy condition is less than 2 percent of the time that the system operates. The procedure changes do not result in a deviation from the guidance in the BTP MEB 3-1 and re-evaluation of the AF system for high-energy pipe breaks is not required. 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.

Based on the above, 50.59 Evaluation 6G-19-005/BRW-E-2019-32 concluded that the Activity can 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-19-005** **BRW-E-2019-32**	**Rev.**	**0** **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 625275** **Revision Number:** **003**

Title: **Replacement of System Aux. Transformer (SAT) 242-2 with ABB Transformer Unit 2**

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, EC 625275, will replace the existing Byron Unit 2 System Auxiliary Transformer (SAT) 242-2 (2AP71E) ASEA 345 kV to 6.9/4.16kV – 67.2 MVA (at 65°C) with a new ABB transformer rated at 345 kV to 6.9/4.16 kV – 67.2 MVA (at 65°C). In addition, a new storage pad for interim storage of the SAT will be constructed in conjunction with this Activity. This temporary storage pad will be removed following the new SAT 242-2 installation. The existing deluge piping will be modified to accommodate the replacement SAT, however, this change is addressed in a separate Activity (EC 625276) and is not addressed in this 50.59 Screening. The EC 625275 activities will also include wiring terminations at the transformer control panel, re-installation of a Serveron analyzer and installation of a new electrical Sudden Pressure Relay (SPR) 2-out-of-3 transducer based logic to replace the existing mechanical 1-out-of-1 SPR logic. In addition, the transformer is provided with a transformer monitoring system (TMS) which monitors and collects various transformer parameters that are displayed locally at the transformer. The transformer is also equipped with low voltage (LV) and high voltage (HV) bushing monitors and a partial discharge (PD) analyzer.

Revision 003 to EC 625275 transfers configuration control of the Unit 2 Open Phase Detection (OPD) system from Clearance Order 02-AP-2PA55J-001 and under the provisions of 10CFR50.65(a)(4) in direct support of maintenance to the EC process. This revision to EC 625275 revises station drawings to reflect the interim/temporary condition with the test switches for the automatic actuation function of the system disabled for both SAT 242-1 and 242-2. Revisions to Calculations BYR13-176, BYR13-177, and BYR13-221 are incorporated under Revision 003 to EC 625275. Revision 1 to 50.59 Screening 6E-18-045 and Revision 0 to 50.59 Evaluation 6G-19-006 provide the justification for this configuration beyond the provisions of 10CFR50.65(a)(4). Revision 003 to EC 625275 will use Clearance Order 02-AP-2PA55J-001 to control and tag the positions of the test switches that are required to be temporarily open by this EC revision. Therefore, Clearance Order 02-AP-2PA55J-001 will remain in place beyond the 90 day limitation for temporary configuration changes in direct support of maintenance, and this Clearance Order will be used as the mechanism for control and tagging of the temporary configuration change.

Note: The configuration change for the Fire Detection and Suppression system supporting the installation of the new SAT 242-2 is accomplished under a separate EC (625276) and is not under the scope of this 50.59 screening.

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

On July 7, 2018, Byron Unit 2 experienced a loss of Bus 13 with a sudden pressure actuation and differential relay actuation on SAT 242-2. External inspection of SAT 242-2 identified a cracked insulator on the A phase high side bushing, leaking from the C phase, and movement of each of the high side bushings. The extent of the damage required the replacement of the transformer.

Revision 003 to EC 625275 documents the interim/temporary condition of the Unit 2 OPD system with its automatic actuation function disabled as a result of the incompatibility of the system with the new SAT 242-2 transformer replaced under this EC. Disabling the trip function eliminates the concern for spurious actuation of the OPD system which would result in the lockout of both SATs on Unit 2.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 625275 **Revision Number:** 003

Title: Replacement of System Aux. Transformer (SAT) 242-2 with ABB Transformer Unit 2

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

The replacement ABB transformer will continue to provide the required first source of offsite power to the 4160V Engineered Safety Bus (ESF) 242 and a source of power to supply non-safety related 6900V Bus 258, and serve as the second source of offsite power to the other unit. The replacement transformer has been procured to have a similar design and the same ratings and addition features as noted below:

- The replacement transformer rating has the same MVA rating as noted below:
  Existing: HV(H) – 40.3/53.8/67.2 MVA, LV1(X) – 25.5/34.1/42.6 MVA, LV2(Y) 14.8/19.7/24.6 MVA (OA/FA/FA @65^{0}C Rise)
  Replacement: HV(H) – 40.3/53.8/67.2 MVA, LV1(X) – 25.5/34.1/42.6 MVA, LV2(Y) – 14.8/19.7/24.6 MVA (ONAN/ONAF/ONAF @65^{0}C Rise)
- The replacement SAT is impedance matched on the existing transformer 36 MVA base as noted below:
  Existing: ZH-X (11.70%), ZH-Y (17.00%), Z-XY (31.90%)
  Replacement: ZH-X (11.94%), ZH-Y (16.42%), Z-XY (31.10%) Note: These are ABB Guaranteed Values.
- The six high voltage side CTs 600/5 A (C-800) and three high voltage side CTs 3000/5 A (C-800) have the same ratios as the existing CTs on the existing SAT 242-2.
- The 2 multi ratio (MR) bushing current transformer (BCTs) furnished by ABB for the replacement SAT 242-2 have the same ratios as the existing CTs on the existing SAT 242-2.
- The existing transformer cooling fans are fed from the 480 VAC Switchgear 234Y Cub. 3B (Normal Feed) and 480 VAC Switchgear 233Y Cub. 2B (Reserve Feed) via an automatic transfer switch at the transformer. The existing SAT cooling system power requirement of the transformer is 12.0 kW, while the replacement SAT cooling system power requirement is 14.0 kW. The transformer auxiliaries draw more than the existing transformer auxiliaries. As documented in EC 625275, due to the increased current on the 480 V switchgear breakers, relay settings were evaluated and changes to breaker settings will be required in conjunction with this Activity.
- As an additional feature, the replacement transformer has a digital transformer monitoring system (TMS). The TMS is located at the transformer and is used to monitor, collect, and display locally, the transformer analog and digital signals associated with various transformer monitored parameters. The new monitoring and control system provides for the operation of the transformer cooling systems.
- As an additional feature, the replacement transformer is provided with an online bushing monitoring system to detect deterioration in the bushings, finding abnormalities in the insulation and issuing actionable alerts.
- As an additional feature, the replacement transformer is provided with an online partial discharge (PD) analyzer to provide actionable alerts when insulation deterioration is detected.
- The proposed modification includes changing the sudden pressure relay logic for the Unit 2 SAT 242-2 from a 1-out-of-1 mechanical SPR logic to 2-out-of-3 electrical SPR logic to improve the functionality of the of the SPR logic. The modified SPR logic uses a two out of three logic scheme which will improve the functional capability of the system by maintaining the capability of the SPR logic to prevent spurious or inadvertent actuation of the transformer lockout trip, even if one SPR relay is not operational, because at least two SPR circuits are required to generate a trip signal. The overall function of the sudden pressure relays is not being changed.
- A review has been performed and has determined that the additional 125 VDC loads associated with the transformer replacement will not have an adverse impact on the DC system battery loading, voltage drop or duty cycle.
- Due to differences in the transformer overall dimensions, a new NSBD termination enclosure is being provided.
- The replacement transformer will be grounded to the existing plant grounding system similar to the ASEA transformer.
- The replacement transformer will be connected to the existing Serveron on-line transformer oil analyzer.
- The ⅛" OD stainless steel tubing and associated tube supports connecting the Serveron helium tank to the helium dryer will be re-worked to accommodate the replacement transformer configuration.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 625275 **Revision Number:** 003

Title: Replacement of System Aux. Transformer (SAT) 242-2 with ABB Transformer Unit 2

- The existing ASEA 242-2 (in Fire Zone 18.10E-2) contains 10,900 gallons of insulating oil per the transformer nameplate. The replacement ABB transformer contains 10,145 gallons of oil. This value bounds the volume assessed for the existing SAT volume. Therefore, EC 625275 has determined that the existing containment enclosure size and drainage system is not adversely impacted by this Activity.
- The existing SAT 242-2 is supported on a concrete foundation, which is a seismic Category II structure. The foundation has been evaluated for placement of the replacement transformer in EC 625275 considering weight, wind, seismic, groundwater pressure, flood pressure and lateral soil pressure loadings. The results of this analysis has concluded that the foundation is adequate for the replacement transformer.
- As an additional feature, the replacement transformer is provided with an online Load Tap Changer (LTC) for future use.

Revision 003 to EC 625275 disables the automatic trip function of the ESF 4KV Bus SAT Feed Breaker upon detection of an open phase condition as described in UFSAR Section 8.3.1.1.2.1. In addition, it incorporates calculation revisions which provide the analytical basis for future revision to the OPD system detection relay setpoints.

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 describes the function of the normally energized SATs is to provide offsite power to the engineered safety features (ESF) 4160-volt bus of the unit. Each SAT also serves as a second source of offsite power for the corresponding ESF bus of the other unit. It has been determined that the replacement Unit 2 SAT 242-2 will accomplish thjs design function.

It is not anticipated that a grid stability study will need to be perform by ComEd or PJM. ATI 4200191-11 is tracking the confirmation that no grid stability study is required for the replacement of SAT 242-2

Unit 2 SAT and 4.16 kV and 6.9 kV protective relay calculations have been revised in conjunction with this Activity. These analyses have determined that relay setting changes are not required except for the OPD relays. There is no prescribed method of evaluation in the UFSAR for performing these evaluations, the use of normally acceptable practices were used and are acceptable.

The physical changes to electrical connections to the non-segregated bus duct (NSBD) system and the transmission system were reviewed and determined to have no impact on the performance of the transformers, NSBD system or switchyard operations.

In conjunction with this activity, an assessment in EC 625275 was performed to address the impact of the SAT temporary storage pad on the probable maximum precipitation (PMP) level. The results of this assessment concluded that the temporary storage pad would not cause the PMP water level to exceed the 870.83 feet value in the UFSAR.

The existing SAT 242-2 is supported on a concrete foundation, which is a seismic Category II structure. The foundation has been evaluated for placement of the replacement transformer in EC 625275 considering weight, wind, seismic, groundwater pressure, flood pressure and lateral soil pressure loadings. The results of this analysis has concluded that the foundation is adequate for the replacement transformer. Note, UFSAR Section 3.3.1.1 states that for Category II structures, a design wind velocity of 75 mph is used. The EC 625275 supporting analysis described above was based on a wind velocity that bounds the above defined UFSAR criteria.

This activity changes the sudden pressure relay logic for the Unit 2 SAT 242-2 from a 1-out-of-1 mechanical SPR logic to 2-out-of-3 electrical SPR logic to improve the functionality of the of the SPR logic. It has been concluded that the change in SPR logic does not involve a change to the logic or operation of the SPR system that adversely affects an UFSAR described design function.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 625275 **Revision Number: 003**

Title: Replacement of System Aux. Transformer (SAT) 242-2 with ABB Transformer Unit 2

The new SPR logic system uses 125 V safety related DC power, as does the existing system. The increase in power requirements for the new system were evaluated and it was determined that there was no adverse impact to the DC voltage drop, battery loading or battery duty cycle.

The new ABB replacement SAT cooling system will utilize the existing Non Class 1E - 480 VAC power sources used for the existing ASEA SAT cooling systems. However, the existing SAT cooling system power requirement of the transformer is 12.0 kW, while the replacement SAT cooling system power requirement is 14.0 kW. The transformer auxiliaries draw more than the existing transformer auxiliaries. As documented in EC 625275, due to the increased current on the 480 V switchgear breakers, relay settings were evaluated and changes to breaker settings will be required in conjunction with this Activity. These breaker setting changes will ensure there is no adverse impact on the auxiliary power system due to the new transformer auxiliary loads. The existing power cables being used to feed the cooling system have been reviewed and found to be acceptable.

The ABB replacement transformer is provided with a transformer monitoring system (TMS) which monitors various analog and digital signals and provides local display and remote communication of select signals to the Unit 2 Computer Room and Plant Process Computer (PPC) as well as initiates alarms in the main control room. A failure of the TMS may cause a loss of transformer performance data from being indicated, or failure of alarm function, or a spurious alarm to occur in the main control room. None of these failures will affect the operation of the transformer.

Additionally, the TMS is equipped with digital controls for the operation of the transformer cooling systems. The design of the cooling system is similar to the existing system to provide air cooled oil cooling. Two groups of cooling are provided and are actuated by the TMS digital control system at defined winding and oil temperature setpoints. Transformer cooling performance could be adversely impacted or loss of the SAT could result from a failure of the TMS digital controls to control the cooling systems. However, as described in UFSAR Section 8.3.1.1.2.1, failure of one of the unit's SATs is already assumed in the offsite power system design. Upon failure of one SAT, removable links can be relocated to connect to the other SAT to supply both divisions. Therefore, failure of the TMS to control the operation of the SAT cooling systems, is already bounded by the loss of the SAT assumed in the above described UFSAR design basis. Additionally, since only one of the Unit 2 SATs with a TMS digital controlled cooling system will result after implementation of this Activity, a common cause software failure of a digitally controlled SAT cooling system is not applicable.

The design bases of the SATs as described in the UFSAR is that each pair of the SATs is sized to provide the total auxiliary power for one unit plus the ESF auxiliary power for the other unit. The replacement SAT 242-2 has the same rating and therefore does not create an adverse change to an UFSAR described design function.

While this Activity will require changes to Operations and Maintenance procedures to address the SAT 242-2 configuration change, the change does not affect UFSAR-described design functions. The annunciation changes implemented in conjunction with this Activity do not impact any information readouts provided to operations to support manual safety functions. This Activity improves the reliability of the SAT 242-2 SPR logic, which is not specifically discussed in the UFSAR.

All evaluations and applicable calculations have been performed in a manner consistent with standard industry practices, existing station design criteria, and evaluation requirements / methodologies in the USAR. No analysis or calculation revision performed for this Activity introduces an adverse change to an element of a USAR described evaluation methodology or uses an alternative evaluation methodology that is used in establishing the design bases or used in the safety analyses. The function of the transformer and the method of performing its function as described in UFSAR Chapter 8 will not change with the installation of the replacement transformer.

Installation and post-modification testing will not subject any SSCs to parameters beyond their design capability. Therefore, the proposed activity does not involve a test or experiment not described in the UFSAR, where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the UFSAR.

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 625275 **Revision Number:** 003

Title: Replacement of System Aux. Transformer (SAT) 242-2 with ABB Transformer Unit 2

No parameters of the SAT or its auxiliary components are listed in the Technical Specifications or the Operating License. Therefore, the proposed activity does not require a change in the Technical Specifications or Operating License.

For the above reasons, and as indicated in the attached 10 CFR 50.59 Screening, a full safety evaluation is not required for the configuration change activities associated with the replacement of SAT 242-2.

The following conclusions are applicable to Revision 003 to EC 625275 as documented in Revision 1 to 50.50 Screening 6E-18-045:

Disabling the automatic trip functions for the Unit 2 OPD system as a result of the system's incompatibility with the SAT 242-2 transformer replaced under EC 625275 represents an adverse change to the UFSAR Section 8.3.1.1.2.1 described design function for the OPD system. UFSAR Section 8.3.1.1.2.1 indicates that upon detection of an open phase condition on the feed to the SAT, the SAT feed breaker to the ESF 4KV busses automatically trips open. All other aspects of the changes incorporated under Revision 003 to EC 625275 do not adversely impact UFSAR described design functions.

No procedure changes are incorporated under Revision 3 to EC 625275 that would adversely impact how UFSAR described design functions are performed or controlled.

The calculation revisions incorporated under Revision 003 to EC 625275 that provide the technical basis for OPD system relay setting changes, do not involve a change in methodology associated with determining the operational setpoints for protective relaying as described in the UFSAR.

The changes incorporated under Revision 003 to EC 625275 do not involve a test or experiment.

The interim/temporary condition of the Unit 2 OPD system with automatic trip functions disabled, does not require a change to the Technical Specifications or Facility Operating License. The OPD instrumentation is not addressed in any Technical Specification.

The following conclusions are applicable to Revision 003 to EC 625275 as documented in 50.59 Evaluation 6G-19-006: With the OPD system automatic actuation functions disabled under Revision 003 to EC 625275, the concern related to the spurious actuation of the system due to its incompatibility with the replacement SAT 242-2 transformer is eliminated. The frequency of occurrence of transients or accidents evaluated in the UFSAR is not changed.

The OPD system does not affect the design functions associated with the loss of power instrumentation addressed under Technical Specification 3.3.5. The compensatory actions in place to support the performance of the design function for the OPD system with it disabled do not replace the automatic functions addressed in Technical Specification 3.3.5. There is no time dependency associated with the likelihood of occurrence for SSC malfunction and the duration that the OPD system is disabled. Existing station procedures do not specify an allowed outage time for the OPD system and the OPD system is not addressed in Technical Specifications.

Dose analyses for UFSAR evaluated accidents or equipment malfunctions do not credit or consider the function of the OPD system. As such, the radiological consequences of accidents or malfunctions are not affected.

The OPD system is considered an enhancement to the preferred offsite power source to protect the Class 1E distribution system from a condition that is outside the original design and licensing of the plant. Since the OPD system is outside the original design and licensing basis for the plant and since the system does not interfere with existing equipment capabilities and equipment protection, disabling the automatic trips does not create the possibility of an accident of a different type than previously evaluated in the UFSAR.

A functional OPD system does not affect the protection provided by the Loss of Power instrumentation addressed in Technical Specification 3.3.5 to trip the SAT feed breakers when a degraded or loss of offsite power occurs. With the system disabled due to its incompatibility with the new SAT 242-2 transformer, transients are eliminated which could challenge the

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: EC 625275 **Revision Number:** 003

Title: Replacement of System Aux. Transformer (SAT) 242-2 with ABB Transformer Unit 2

plant. The duration in which the OPD system automatic trips are disabled beyond the provisions of 10CFR50.65(a)(4) does not create the possibility for accidents of different type than previously evaluated in the UFSAR.

No reanalysis of any existing safety analysis is required to extend the duration that the OPD system automatic trips are disabled. Therefore, all prior results remain valid and demonstrate that design basis limits for the fission product barriers are not exceeded for postulated events. In addition, the proposed activity does not interfere with existing equipment capabilities and equipment protection. 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.

In summary, transferring the configuration control of the Unit 2 OPD system to EC 625275 to address the impacts on the system due to the replacement of SAT 242-2 and maintaining Clearance Order 02-AP-2PA55J-001 in place beyond 90 days does not require 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-18-045**	**Rev.**	**1**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-19-006**	**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): Byron Unit 2

Activity/Document Number: EC 630528 **Revision Number:** 1

Title: Lost Parts Evaluation for 2B Charging Pump Mechanical Seal Debris

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 disassembly of the 2B CV charging pump inboard seal under WO 4942198, it was discovered that the mechanical seal had catastrophically failed. Debris from the failed seal was found in the seal cooling piping which mixes with the process fluid in the equalization line mixing chamber and has a direct path to the pump suction. Both seal injection pressure differential (DP) trends and chemistry trends of tungsten (one of the materials comprising the mechanical seal) in the Reactor Coolant System (RCS) indicate that the debris was likely transported through the CV pump and distributed in the RCS. The exact quantity and size of material is unknown; however, the seal housing was weighed before and after rebuild and the difference in weight was approximately 1.4 lbs. Thus 1.4 lbs is a conservative estimate of the amount of material lost in the RCS. Per EC 630528, the maximum particle size that could be transported to the pump suction is approximately 1/8 inch. The debris cannot be fully removed from the RCS. Thus, the presence of the debris within the RCS is treated as a degraded condition which will be accepted as-is. The impact of the lost material on the RCS and primary system components was evaluated in EC 630528. Due to the accept as-is disposition, this activity is subject to evaluation under 10 CFR 50.59.

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

As discussed above, the debris cannot be fully removed from the RCS. Therefore, it is being accepted as-is and evaluated under 10 CFR 50.59.

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

The presence of the debris in the RCS has been evaluated in EC 630528. It has been concluded that the presence of the debris will not significantly affect the RCS or any primary system components or connected systems. Thus, the proposed activity does not significantly 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 evaluations performed in EC 630528 considered the following accidents:

- Stuck Control Rod (UFSAR Section 4.3, Section 15.4.3)

An evaluation of the lost material and effect on the control rods was performed by Nuclear Fuels Services as documented in EC 630528. The lost material does not introduce the possibility of a change in the frequency of an accident because the lost material is not an initiator of any accident and does not create new failure modes. Per EC 630528, the debris particles are too small to cause obstruction of a Rod Cluster Control Assembly (RCCA). Therefore, there is no impact with respect to RCCA interference.

- Locked Rotor (UFSAR Section 15.3.3), Decrease in RCS Forced Flow (UFSAR Section 15.3), and RCP Failure / Small Break Loss of Coolant Accident (UFSAR Section 15.6.5)

An evaluation of the lost material with regard to the reactor coolant pumps was performed in EC 630528. The debris does not introduce the possibility of a change in the frequency of an accident because the lost material is not an initiator of any accident and does not create new failure modes. As evaluated in EC 630528, the debris is expected to pass through the pump hydraulic passages with no change in pump vibrational characteristics or increase in locked rotor probability due to the small size of the debris particles. The debris is not capable of causing a significant mechanical impact due the small size of the debris particles.

Under normal operation, the debris will not enter seal injection as it would be filtered out by the seal injection filters. However,

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in certain accident scenarios, seal injection flow is lost, allowing RC to flow past the No. 1 seal. Coolant is prevented from further travel by the safe shutdown seals, which are located between the No. 1 and No. 2 seals. If the debris were to reach the seals, all the very smallest particles would not reach the safe shutdown seals due to the 10-micron (approximately 0.0004 inch) gap between the faces of the No. 1 seal. The particles less than 10 micron that could pass through this gap are not large or heavy enough to cause damage to the safe shutdown seals or No. 2 and No. 3 seals by engineering judgment.

Therefore, there is no impact with respect to RCS forced flow, RCP failure, locked rotor, or small break loss of coolant accident.

- Steam Generator Tube Leak / Rupture (UFSAR Section 15.6.3)

For the purposes of foreign material evaluation, the Byron U2 steam generator is comparable to the steam generators evaluated in the Zion loose parts evaluation referenced in EC 630528. Therefore, the evaluation of the Zion steam generators is applicable to Byron Unit 2. The impact of the debris on the tube sheet, divider plate, channel head, tube ends, and primary side welds during operation of the steam generator was evaluated for the Zion generators. The Byron Unit 2 steam generator tube sheet thickness is 21.03 inches. The tubes have a nominal outside diameter of 0.750 inches with a minimum wall thickness of 0.043inches. Due to the small size and mass of the debris generated by the 2B charging pump mechanical seal, no adverse effects on the integrity of the Byron Unit 2 steam generators are expected. The debris will pass through the tubes without impact any significant impact energy. There is not potential to plug tubes because of the small size of the particles, limited amount of debris in the RCS, and dispersion throughout the large RCS volume. Therefore, there is no impact with respect to the steam generators.

Therefore, based on the above evaluations as presented in EC 630528, there is not more than a minimal increase in the frequency of occurrence of an accident previously evaluation in the UFSAR.

The evaluation performed in EC 630528 concluded that the debris would not have an adverse effect on the primary system components. The debris particles are very small and do not possess the mass, volume, or area to cause significant operational concerns as they flow through the above components and systems. The debris does not introduce the possibility of a change in likelihood of an equipment malfunction because the lost material is not an initiator of any accident and does not create new failure modes. The effect on nuclear fuel was also evaluated in EC 630528. Due to the small size and geometry of the debris, they are expected to travel through the fuel nozzle channels without causing blockage or posing a significant fretting concern. Therefore, based on the above evaluation as presented in EC 630528, there is not 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 debris does not introduce the possibility of a change in the consequences of an accident because the lost material is not an initiator of any accident and does not create new failure modes. The debris was evaluated for impact on current safety analyses, the RCS, primary system components, fuel, and connected systems. It was determined that they would not be adversely affected. As discussed in EC 630528, the debris does not pose a significant fretting concern. In the unlikely event that fretting does occur, this would not in itself increase the consequences of an accident previously evaluated because UFSAR Section 4.2 states "It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the plant cleanup system and are consistent with plant design bases. The number of rod failures is small enough such that the dose limits given in 10CFR100 and 10CFR50.67 will not be exceeded." Therefore, there is not more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

The debris does not introduce the possibility of a change in the consequences of a malfunction because the because the lost material is not an initiator of any accident and does not create new failure modes. The debris was evaluated for impact on current safety analyses, the RCS, primary system components, fuel, and connected systems. It was determined that they would not be adversely affected. As discussed in EC 630528, the debris does not pose a significant fretting concern. As stated in the response to Question 3 above, even an unlikely fretting event would be bounded by the UFSAR discussion in Section 4.2. Due to the limited amount of debris, the potential for fretting to result in multiple fuel rod failures is extremely low. Thus, it is extremely unlikely there would be multiple fuel failures. Furthermore, the primary system components, as evaluated in EC 630528, would not malfunction in a manner different than previously evaluated in the UFSAR. Therefore, there is not more than minimal increase in the consequences of a malfunction of an SSC important to safety.

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The debris does not introduce the possibility of a new accident because the lost material is not an initiator of any accident and does not create new failure modes. The debris was evaluated in EC 630528 for impact on current safety analyses, the RCS, primary system components, fuel, and connected systems. It was determined that any affected component or system would not malfunction in a manner different than previously evaluated in the UFSAR. Therefore, the proposed activity does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR.

The debris does not introduce the possibility for a malfunction of an SSC with a different result because the lost material is not an initiator of any accident and does not create a failure mode that is not bounded by the UFSAR. The components and systems potentially impacted by the debris have been evaluated in EC 630528. It was determined that these components and systems would continue to perform their design functions. As discussed above, fuel cladding damage is already assessed in UFSAR Section 4.2. A comparison of the identified failure modes indicates that the result of the failure modes resulting from the lost material are bounded by those presented 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.

The debris does not result in a change that would cause any system parameter to change. As evaluated in Questions 2 and 3, the potential for limited fuel fretting caused by the debris is within the plant's cleanup system capacity and is consistent with the design basis as discussed in UFSAR Section 4.2. As discussed in Question 4 above, there is not enough debris to cause widespread fretting of multiple fuel rods. Therefore, the lost material does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

The evaluation of the debris per EC 630528 does not involve a method of evaluation as defined in the UFSAR and does not use an alternative method of evaluation for UFSAR-described safety analyses. Therefore, the evaluation of the debris 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 debris was evaluated in EC 630528 for impact on current safety analyses, the RCS, primary system components, and connected systems. It was determined that these components and systems would continue to perform their design functions. As such, the Technical Specification for these systems and components are not affected by the presence of the debris in the RCS. Neither the Technical Specifications nor the Facility Operating License prohibit the presence of the subject debris in the RCS. Therefore, the proposed activity does not require a change to the Technical Specifications or Facility Operating License.

Therefore, the proposed activity may be implemented per plant procedures and a license amendment is not required.

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-20-001	**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.