Exelon Generation.

Byron Generating Station

4450 North German Church Rd
Byron, IL 61010-9794

www.exeloncorp.com

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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: Report of Changes, Tests and Experiments

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, 2014 through November 30, 2016 and consists of 50.59 Review Coversheets for changes to the facility or procedures as described in the Updated Final Analysis Report (UFSAR) and test or experiments not described in the UFSAR.

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

Respectfully,

Mark E. Kanavos
Site Vice President
Byron Generating Station

MEK/LZ/sg

cc: NRC Regional Administrator – NRC Region III

Enclosure: Byron Station 10 CFR 50.59 Review Coversheets for January 1, 2014 through November 30, 2016

Byron Station 10 CFR 50.59 Report

10 CFR 50.59 Review Coversheets
for
January 1, 2014 through November 30, 2016

Tracking No.	Revision	Number or Identifier	Subject or Title
6G-13-013	0	DRP 15-029	Misload Fuel Assembly Analysis supporting DRP 15-029
6G-13-016	0	DRP 15-059	Byron Station Polar Crane Single Failure-Proof Equivalency per NEI 08-05
6G-14-001	0	DRP 15-067 / EC 389969	Unrestrained HI-TRAC/HI-STORM Stackup Configuration in the FHB
6G-14-003	0	DRP 15-107 / EC 396144	Fuel Rod Design Changes
6G-15-002	0	EC 400682 and EC 401062	Unit 1 & Unit 2 Reactor Vessel Head CRDM Nozzle Water Jet Peening
6G-15-004		EC 401137	Unit 1 RCP Safe Shutdown Seal
6G-15-005	0	EC 362147, 362149, 362150, 362152, 362151, 362153, 362146, 362148	Instrument Inverter Replacements 111, 112, 113, 114, 211, 212, 213, 214
6G-15-006	0	EC 402345 (Unit 2)	Replace RCP Underfrequency KF Relays with ABB Circuit Shield Type 81 Frequency Relays to Address ABB Part 21 Notification of Potential Defect For KF Relay ZPA
6G-16-001	0	EC 404997	Remove AF Diesel Air Intake Elbow and Blank Off TB Air Intake
6G-16-002	0	EC 400277	Process Revised AST Calculations
6G-16-003	0	AT 01493278-02	Evaluation of Long-Term Removal from Service of the CVCS Positive Displacement Pump to Address 2013 NRC Inspection Finding at BWD
6G-16-006	0	EC 406220	Reroute AF Diesel Combustion Aire Intake to 364' General Area

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: DRP 15-029, EC 392455, EC 392431, EC 391626, EC 392442, NF-BY-312, BwVS TRM 3.1.H.1
Revision Number: 0, 0, 0, 1, 0, 8, 17

Title: Implementation of WCAP-16676-NP, "Analysis Update for the Inadvertent Loading Event"

OTE: 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 is the implementation of an updated generic analysis of the Inadvertent Loading of a Fuel Assembly into an Improper Location (aka Misload) event, described in UFSAR Section 15.4.7. Implementing this generic analysis will require an update to UFSAR Section 15.4.7 to revise the information contained within this section, adding a reference to Section 15.4, and eliminating Figures 15.4-13 through 15.4-17. Revisions to NF-BY-312 and BwVS TRM 3.1.H.1 will also be required to include the flux map review criteria associated with this generic analysis.

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

UFSAR Section 15.4.7 currently describes the analysis of the Misload event as a comparison of the X-Y power distributions for a properly loaded core to the X-Y power distributions for a case with a misload configuration. The cases that were previously analyzed and described in the UFSAR are as follows:

Case A: Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered was the interchange to two adjacent assemblies near the periphery of the core.

Case B: Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. Two analyses have been performed for this case:

In Case B-1, the interchange is assumed to take place with the burnable poison rods transferred with the Region 2 assembly mistakenly loaded into Region 1.

In Case B-2, the interchange is assumed to take place closer to core center and with burnable poison rods located in the correct Region 2 position but in a Region 1 assembly mistakenly loaded in the Region 2 position.

Case C: Enrichment error: Case in which a Region 2 fuel assembly is loaded in the core central location.

Case D: Case in which a Region 2 fuel assembly instead of a Region 1 fuel assembly is loaded near the core periphery.

The enrichments for the fuel regions utilized in the original analysis were 2.10 w/o U-235 for Region 1, 2.60 w/o U-235 for Region 2, and 3.10 w/o U-235 for Region 3. While these were typical enrichments at the time, modern core designs utilize much higher enrichments, typically in the 4.20-4.95 w/o U-235 range. In addition, the types of burnable absorbers used in the core are different and loading pattern strategies have changed since the previous analysis was performed, notably in the fact that fresh fuel is no longer loaded on the periphery of the core but is instead interspersed with once-burned assemblies on the interior and once- or twice-burned fuel is placed on the periphery. In addition, reload batch sizes have increased from 64-65 assemblies to 88-89 assemblies or more. Based on these factors, the previous analysis that was discussed in UFSAR Section 15.4.7 doesn't represent current core designs.

In 2001, Byron and Braidwood revised the power level at which the first flux map could be taken from <30% rated thermal power (RTP) to up to <50% RTP, based on input from Nuclear Fuels. To support this effort, Nuclear Fuels has been performing cycle-specific design analyses in order to confirm whether delaying the first flux map to ~50% RTP is acceptable. The cycle-specific analyses used the cases from the UFSAR as a guideline as to what scenarios needed to be evaluated and took into account current core design practices, enrichments, and burnable absorbers. The cycle-specific analyses confirmed that the current flux map review criteria were similarly capable of detecting the misload scenarios described in the UFSAR at both 30% RTP and 50% RTP and, if they were not, established new acceptance criteria for that specific beginning of cycle flux map.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: DRP 15-029, EC 392455, EC 392431, EC 391626, EC 392442, NF-BY-312, BwVS TRM 3.1.H.1
Revision Number: 0, 0, 0, 1, 0, 8, 17

Title: Implementation of WCAP-16676-NP, "Analysis Update for the Inadvertent Loading Event"

These cycle-specific analyses also determined whether the $F_{\Delta H}$ values resulting from the misload scenarios exceeded the RTP Core Operating Limits Report (COLR) $F_{\Delta H}$ limit adjusted for off-rated conditions so that it could be confirmed that the acceptance criteria for this event as described in NUREG-0800, the Standard Review Plan (SRP), Section 15.4.7, would still be met. The acceptance criteria for this event, as described in SRP Section 15.4.7, Rev. 1 (issued in 1981) are:

a. To meet the requirements of [General Design Criterion] 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel loading errors after fueling operations.
b. In the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 guidelines.

The cycle-specific analyses performed by Nuclear Fuels confirmed that <1% of the fuel would have $F_{\Delta H}$ values in excess of the RTP COLR $F_{\Delta H}$ limit adjusted for off-rated conditions. The 1% criterion is consistent with the assumptions documented in UFSAR Section 12.2, Radiation Sources, since it was conservatively assumed that a fuel rod with an $F_{\Delta H}$ value in excess of this value would be likely to fail, when in reality, only rods with an $F_{\Delta H}$ value above the limit for DNB at normal operating conditions would be likely to fail.

Westinghouse performed an updated generic analysis for this event as described in WCAP-16676-NP. The Westinghouse updated analysis was performed using more representative enrichments (~4.00 - ~4.80 w/o U-235), updated burnable absorber types, current loading pattern strategies, and a more representative batch size (84 assemblies). This analysis also looked at the difference in detectability not only between 30% RTP and 50% RTP, but also depending on the number of detector locations available during the flux map. Consistent with the original generic analysis, Westinghouse performed the analysis by comparing the X-Y power distributions for cores with and without misloads to determine the power distribution impacts of misloaded assemblies. A calculation of the resulting $F_{\Delta H}$ values was also performed to determine whether the resulting $F_{\Delta H}$ values would exceed the RTP COLR limit or the limit for Deptarture from Nucleate Boiling (DNB) at normal operating conditions. Westinghouse assumed a RTP COLR $F_{\Delta H}$ limit of 1.65 and a limit for DNB at normal operating conditions of 1.99 for these comparisons. (In comparison, the Byron and Braidwood RTP COLR $F_{\Delta H}$ limit is 1.70, and the limit for DNB at normal operating conditions is 2.16 for non-MUR conditions and 2.14 for MUR conditions. For $F_{\Delta H}$ lower limits are more restrictive, so the limits assumed in the updated generic analysis are conservative with respect to the limits for Byron and Braidwood.)

A series of 500 random assembly swaps, encompassing the types of misload scenarios that are currently detailed in the UFSAR, were utilized in the updated analysis along with 1000 combinations of unavailable detectors. In addition, a series of 1158 single assembly "misloads" were analyzed with 1000 combinations of unavailable detectors, covering a range of reactivities from a very highly reactive feed assembly to a low reactivity twice-burned assembly. Of the 500 assembly swap configurations that were analyzed, only 66 were shown to result in an $F_{\Delta H}$ in excess of the limit for DNB at normal operating conditions, and of the 1158 single assembly misloads analyzed, only 242 were shown to result in an $F_{\Delta H}$ in excess of the limit for DNB at normal operating conditions. The Westinghouse analysis described in WCAP-16676-NP is not only more comprehensive than both the original generic analysis and the cycle-specific analyses that Nuclear Fuels has been performing by evaluating several hundred misload scenarios instead of the five scenarios analyzed in the original generic analysis, but the WCAP also establishes a set of review criteria for flux maps. The current flux map review criteria used for the initial cycle flux map are based on a recommendation in ANSI/ANS 19.6.1-2011, "Reload Startup Physics Tests for Pressurized Water Reactors." The criteria established by Westinghouse as part of this analysis are not only more stringent than the current criteria based on the addition of a comparison of the measurements in symmetric locations in addition to the current comparison of the measured-to-predicted reaction rates, but also take into account the effects of having detectors out of service. These criteria were shown to be capable of detecting a large fraction (> ~60%) of the misload scenarios that would result in an $F_{\Delta H}$ in excess of the RTP COLR limit and > 99% of the misload scenarios that would result in an $F_{\Delta H}$ in excess of the limit for DNB at normal operating conditions, even if not all detector locations are available for the initial flux map. In addition, this analysis shows that, of the misload scenarios that would result in $F_{\Delta H}$ in excess of the limit for DNB at normal operating conditions, only a small fraction of them (<< 1%) are not capable of being detected during an initial flux map. This satisfies the confirmation that Nuclear Fuels was performing in their cycle-specific analyses that the flux map review criteria would be effective in detecting fuel failures, since the least restrictive criteria established by Westinghouse in this analysis are more restrictive than the criteria currently used by Byron and Braidwood.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: DRP 15-029, EC 392455, EC 392431, EC 391626, EC 392442, NF-BY-312, BwVS TRM 3.1.H.1
Revision Number: 0, 0, 0, 1, 0, 8, 17

Title: Implementation of WCAP-16676-NP, "Analysis Update for the Inadvertent Loading Event"

Westinghouse performed an additional analysis on potentially undetected misload scenarios to determine what fraction of the core would have $F_{\Delta H}$ values in excess of the limit for DNB at normal operating conditions. This additional analysis assumed the maximum number of inoperable detectors in combination with the least restrictive flux map review criteria in order to maximize the number of misload scenarios that would not be detected. Each of the misload cases was simulated in ANC, and full power depletions were performed. Fuel census data was collected from these depletions, which provides the percentage of fuel rods in the core above a given relative power. The best estimate fuel rod powers were increased by an 8% design allowance, which effectively increases the number of fuel rods above a given relative power. Westinghouse determined that, given these constraints, the most limiting time in the cycle would be at the beginning of cycle (150 MWD/MTU). The results of the most limiting misload case at this most limiting time in the cycle showed that, while 1.5%-2.0% of the core would have $F_{\Delta H}$ values above the RTP COLR limit (a percentage which would be decreased to below 1% if the $F_{\Delta H}$ limit was adjusted for off-rated conditions), only 0.58% of the core would have $F_{\Delta H}$ values in excess of the limit for DNB at normal operating conditions. This satisfies the criteria that <1% of the core would have $F_{\Delta H}$ values in excess of the limit for DNB at normal operating conditions that Nuclear Fuels was confirming in their cycle-specific analyses.

As described above, implementing the generic analysis described in WCAP-16676-NP will establish a set of flux map review criteria that have been shown to have the capability of detecting > ~60% of the misload scenarios that would result in an $F_{\Delta H}$ in excess of the RTP COLR limit and > 99% of the misload scenarios that could would result in an $F_{\Delta H}$ in excess of the limit for DNB at normal operating conditions. The newly established flux map review criteria are more restrictive than the criteria currently in use by Byron and Braidwood, and the analysis confirms that, even for the worst undetectable misload scenario, <1% of the core would have $F_{\Delta H}$ values in excess of the limit for DNB at normal operating conditions. It would also eliminate the need for Nuclear Fuels to perform cycle-specific analyses for the Misload event and would update the UFSAR to describe the updated analysis and show the new flux map criteria. Revisions to NF-BY-312 and BwVS TRM 3.1.H.1 will update the flux map review criteria utilized by Byron and Braidwood that were established based on the generic analysis performed by Westinghouse.

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

Implementation of the generic analysis described in WCAP-16676-NP will accomplish several things. First, it will establish a set of flux map review criteria that have been shown to be able to detect > ~60% of the misload scenarios that would result in an $F_{\Delta H}$ in excess of the RTP COLR limit and > 99% of the misload scenarios that would result in an $F_{\Delta H}$ in excess of the limit for DNB at normal operating conditions. Second, the analysis also is much more comprehensive than both the original analysis performed by Westinghouse and the analyses that Nuclear Fuels has been performing. It accounts for current core design practices in addition to validating that the capability of detecting misload scenarios is similar between 30% RTP and 50% RTP, which formally documents that it is acceptable to proceed up to 50% RTP before taking the first flux map of the cycle, so the need for Nuclear Fuels to perform cycle-specific analyses will be eliminated. Third, this implementation will lead to an update to UFSAR Section 15.4.7 that describes the updated analysis and establishes the flux map review criteria. Last, the procedures that are utilized by Byron and Braidwood will be updated with the new flux map review criteria to ensure both sites are using a consistent set of review criteria that are capable of detecting a large fraction of misload scenarios.

Tracking No. 6G-13-013

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: DRP 15-029, EC 392455, EC 392431, EC 391626, EC 392442, NF-BY-312, BwVS TRM 3.1.H.1
Revision Number: 0, 0, 0, 1, 0, 8, 17

Title: Implementation of WCAP-16676-NP, "Analysis Update for the Inadvertent Loading 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 proposed Activity, which will implement the updated generic analysis for the Inadvertent Loading and Operation of a Fuel Assembly in an Improper Location (aka Misload) event as described in WCAP-16676-NP, "Analysis Update for the Inadvertent Loading Event," and update the UFSAR Section 15.4.7 to describe the updated generic analysis as well as revising procedures NF-BY-312 and BwVS TRM 3.1.H.1 in order to implement the flux map review criteria developed as part of the WCAP-16676-NP evaluation, can be performed without NRC review or approval. The attached 50.59 Screening concludes that the proposed Activity does not involve a change to an SSC that adversely affects an UFSAR described design function (Question 1), nor does it 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 in a manner that is inconsistent with the analyses or descriptions in the UFSAR (Question 4) or require a change to the Technical Specifications or Facility Operating License (Question 5). The criteria used for reviewing flux maps, which is being revised in NF-BY-312 and BwVS TRM 3.1.H.1 is not currently described in the UFSAR, and the method of performing flux maps is not being changed. The minimum number of detectors required to be available for an initial flux map is not being revised, so the proposed Activity does not involve a change to a procedure that adversely affects how UFSAR described design functions are performed or controlled (Question 2). UFSAR Section 15.4.7 currently contains a description of the evaluation of the Misload event, but the method used for evaluating the impacts of misload scenarios is the same in the updated generic analysis as it was in the original analysis and the cases that were analyzed are more comprehensive than those used in the original analysis. In addition, no flux map review criteria used to determine whether a misload configuration exists are currently defined in the UFSAR. However, the X-Y power distributions that were calculated for the updated analysis used a different computer code than the codes used in the original analysis, which constitutes the use of an alternative methodology and requires a 50.59 Evaluation (Question 3).

The attached 50.59 Evaluation concluded that the computer code that was used to generate the X-Y power distributions used in this updated generic analysis (ANC) was previously approved by the NRC as a direct replacement for the computer codes that were described in the UFSAR as being used in the original analysis (TURTLE and PALADON). ANC was previously approved for use by Exelon by the NRC, and is already described in the UFSAR as a code used for the calculation of X-Y power distributions. Therefore, use of this code in the generation of X-Y power distributions for the analysis of the Misload event does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses, and thus NRC approval is not required prior to its use in this capacity.

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

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	**6D-13-016/ BRW-S-2013-115**	**Rev.**	**2 / 2**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-13-013 / BRW-E-2013-134**	**Rev.**	**1 / 1**

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: **DRP 15-059** **Revision Number:** **0**

Title: **UFSAR Revision for Reactor Head Lift – Polar Crane Single Failure Proof Equivalency per NEI 08-05**

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 changes the requirements necessary to perform the Reactor Vessel Head lift over irradiated fuel. This change incorporates the guidelines contained in NEI 08-05, "Industry Initiative on Control of Heavy Loads", necessary to declare the Polar Crane "Single Failure-Proof Equivalent" (SFPE). This change provides a SFPE crane for handling the Reactor Vessel Head, thus ensuring the risk of a load drop on irradiated fuel is minimal. This provision is in lieu of a Reactor Head drop analysis. The polar crane is not modified by this change.

Note: This activity is applicable to Byron Units 1 & 2 and Braidwood Unit 2. Braidwood Unit 1 will be addressed under a separate activity.

Reason for Activity:

(Discuss why the proposed activity is being performed.)

The proposed activity is being performed to address an industry issue applicable to Byron and Braidwood Stations with respect to handling of the Reactor Vessel Head over irradiated fuel. This issue is discussed in NEI 08-05, "Industry Initiative on Control of Heavy Loads".

Effect of Activity:

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

This activity revises the methodology used to ensure that the handling of the Reactor Vessel Head over irradiated fuel does not pose a significant risk to the public health and safety. Currently, the methodology used relies on the results of a Reactor Head Drop analysis to ensure that the fuel remains covered and sufficient cooling is available. The proposed activity uses an NRC approved methodology to consider the Polar Crane as SFPE to provide this assurance.

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 changes the requirements necessary to perform the Reactor Vessel Head lift over irradiated fuel. This change incorporates the guidelines contained in NEI 08-05 "Industry Initiative on Control of Heavy Loads", necessary to declare the Polar Crane SFPE. This change provides a SFPE crane for handling the Reactor Vessel Head, thus ensuring that the risk of a load drop on irradiated fuel is minimal. This provision is in lieu of a Reactor Head drop analysis. The polar crane is not modified by this change. Only the procedures utilized for inspection and maintenance of the crane, as well as those utilized for handling the Reactor Head are affected.

Procedures associated with this activity include those utilized for inspection and maintenance of the crane, as well as those utilized for handling the Reactor Head. These procedures contain the necessary instructions to satisfy the SFPE requirements per NEI 08-05; therefore, there are no adverse changes to these procedures.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: **DRP 15-059** **Revision Number:** **0**

Title: **UFSAR Revision for Reactor Head Lift – Polar Crane Single Failure Proof Equivalency per NEI 08-05**

This activity changes the methodology to address handling of the Reactor Vessel Head. Currently, the potential for a load drop involving the Reactor Vessel head is addressed by performance of a load drop analysis. The proposed activity utilizes the guidelines provided in NEI 08-05 to provide a SFPE crane. This methodology has been approved for use by the NRC as stated in Regulatory Issue Summary (RIS) 2008-28 "Endorsement of Nuclear Energy Institute Guidance for Reactor Vessel Head Heavy Load Lifts". A Safety Evaluation accepting this methodology was provided by the NRC in a letter dated September 5, 2008.

There is no test or experiment performed by this activity. The Polar Crane is the only SSC utilized under this activity. Enhancements to the Polar Crane inspection and Reactor Head lifting procedures are provided to minimize the potential for a load drop accident involving the Reactor Head.

There are no Technical Specifications associated with use of the Polar Crane for lifting the Reactor Head. Therefore, no changes to the Technical Specifications or facility Operating License are required.

Therefore, except for Question 3 regarding evaluation methodologies, a 50.59 screening is appropriate for this activity. The change in evaluation methodology is addressed in a 50.59 evaluation (Question 8, only). The answer to this question is "No"; therefore, the proposed change may be implemented without 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-13-202 /**	**Rev.**	**0**
			BRW-S-2013-186	**Rev.**	**0**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-13-016 /**	**Rev.**	**0**
			BRW-E-2013-187	**Rev.**	**0**

Tracking No. 6G-14-001

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Unit 0

Activity/Document Number: DRP 15-067 / EC 389969 / BFP FH-65 / BFP FH-66 / BFP FH-69 / BFP FH-81 / BFP FH-83
Revision Number: 0 / 0 / 12 / 0 / 12 / 3 / 4

Title: Unrestrained HI-TRAC/HI-STORM stack-up configuration in the Fuel Handling Building

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).

Background
The current Dry Cask storage process of loading an MPC into the HI-STORM and the proposed changes are as follows: Prior to the start of this activity, new east and west grillage platforms are staged at the location of the HI-STORM at the MPC transfer location. The center grillage platform is staged south of the MPC transfer location. The HI-STORM (with an empty MPC) is brought into the FHB on a Low Profile Transporter (LPT) by the Trackmobile. The LPT is a cart with rollers that travels along the rails embedded in the FHB trackway floor (EL. 401). The LPT is then connected to the center grillage platform. The HI-STORM is then raised off the LPT using four (4) hydraulic jacks. Using the Trackmobile the LPT is moved out from under the HI-STORM and the center grillage platform is staged directly under the HI-STORM. The HI-STORM is then lowered onto the combined grillage platforms (east, center and west grillage platforms). The combined grillage platforms are at the same elevation (corresponds to the bottom of the HI-STORM). The HI-STORM lid is removed using the FHB overhead crane; the empty MPC is removed from the HI-STORM by the FHB crane and placed inside the HI-TRAC located in the Decontamination Pit. The HI-TRAC is then transported by the FHB crane to the cask loading area of the Spent Fuel Pool (SFP). The new modified Mating Device (MD) is placed on top of the HI-STORM using the FHB overhead crane and is bolted to the top of the HI-STORM prior to being released from the control of the FHB crane. The MPC (inside the HI-TRAC transfer cask) is loaded with spent fuel assemblies from the Spent Fuel Pool (SFP). After fuel loading has been completed the HI-TRAC is transported by the FHB crane to the Decontamination Pit for MPC processing. During MPC processing activities, the inside of the MPC is dried, filled with helium and the MPC lid is welded to the top of the MPC. At this point, the HI-TRAC (with MPC) is lifted by the FHB crane and placed on top of the MD, bolted to the MD, and then released from the control of the FHB crane. The FHB crane then lowers the MPC from the HI-TRAC into the HI-STORM. After the MPC transfer has been completed the HI-TRAC is attached to the FHB crane, unbolted from the MD and transported to the Decontamination Pit.

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

The proposed activity being evaluated is the process of loading an MPC into the HI-TRAC/HI-STORM unrestrained stacked arrangement instead of the existing approach where the stack is equipped with temporary lateral seismic supports. The process begins when the HI-TRAC is being attached to the MD on top of the HI-STORM and ends when the HI-TRAC is back under the control of the FHB crane, after being unbolted from the MD. This includes lowering the MPC into the HI-STORM.
The proposed activity involves the elimination of the existing seismic support structure currently erected in the Fuel Handling Building (FHB) in advance of a dry cask storage campaign and instead using the "unrestrained stack-up" concept. The unrestrained stack-up approach involves the use of grillage support assemblies (GSA) beneath the HI-STORM/HI-TRAC stack to absorb and distribute energy and provide a surface that allows the stack to move in a seismic event without tipping over. In addition, a mating device (MD) is used to fasten the HI-TRAC on top of the HI-STORM container. The combination of these new features would allow elimination of the existing seismic support structure.

The following procedures have been revised to support the implementation of the unrestrained freestanding stack-up configuration: BFP FH-65, 66, 69, 81, and 83

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

In past dry cask storage campaigns, the HI-TRAC and HI-STORM containers were restrained by temporary steel supports/restraints attached to the FHB structure and qualified for seismic loads. The size of these restraints adversely impacted physical access to the FHB; required significant time to erect and remove (~$600k/campaign); required storage in sea vans outside which challenges RP limit for outdoor storage; potentially affects refueling outage preparations or other emergent plant activities that required access to the FHB. An unrestrained freestanding stackup configuration will maximize access to the FHB and reduce costs and accumulated dose associated with dry cask storage.

Tracking No. 6G-14-001

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Unit 0

Activity/Document Number: DRP 15-067 / EC 389969 / BFP FH-65 / BFP FH-66 / BFP FH-69 / BFP FH-81 / BFP FH-83
Revision Number: 0 / 0 / 12 / 0 / 12 / 3 / 4

Title: Unrestrained HI-TRAC/HI-STORM stack-up configuration in the Fuel Handling Building

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

The activity affects the Dry Cask Storage (DCS) system and the current approach to loading a multi-purpose canister into the HI-STORM container as described in UFSAR, Section 9.1.

The activity has the potential to affect the FHB and FHB SSCs adjacent to the unrestrained stack-up configuration; this activity has no impact on normal plant operations. The basic function of the FHB is to provide a Class I structure capable of safely housing/supporting the spent fuel pool, new fuel storage area and portions of the fuel transfer system while withstanding transient load conditions and all credible natural phenomenon (design-basis earthquakes, tornados, and missiles). Supporting analysis has demonstrated that this design change will not adversely impact the basic function of the FHB.

The applicable accidents previously evaluated in the UFSAR are a fuel handling accident (UFSAR Section 15.7.4) and a spent fuel cask drop accident (UFSAR Section 15.7.5).

The fuel handling accident inside the spent fuel storage building (i.e. FHB) is described as dropping of a spent fuel assembly in the spent fuel pool resulting in the rupture of the cladding of fuel rods. The FHB and associated FHB SSCs are relied upon to mitigate the consequences of a fuel handling accident inside the FHB. The unrestrained freestanding stack-up configuration was evaluated for normal operating conditions and during a design basis seismic event by analyses. From the results of the aforementioned analyses it was determined that the FHB and FHB SSCs adjacent to the unrestrained freestanding stack-up configuration will not be adversely impacted because the HI-TRAC and HI-STORM stack-up configuration is stable and will not tip over on or slide into the FHB structure or FHB SSCs adjacent to the stack-up configuration and that the FHB and the components of the unrestrained freestanding stack-up configuration (e.g. HI-TRAC, HI-STORM, Mating Device (MD), connections between MD and HI-TRAC or HI-STORM, grillage platforms) are structurally adequate for all design basis loads. Based on the location of the unrestrained freestanding stack-up configuration in the FHB, it will be physically impossible for the HI-TRAC and/or HI-STORM to fall into the Spent Fuel Pool (SFP) and initiate a fuel handling accident in the SFP. As a result the activity does not affect the safety analyses for the fuel handling accident described in UFSAR Section 15.7.4.

The spent fuel cask drop accident is no longer credible when the HI-TRAC or MPC are under control of the FHB crane because the FHB overhead crane and the associated lifting devices and HI-TRAC and MPC attachment points meet the single failure proof requirements of NUREG-0612, NUREG-0554 and ASME NOG-1-2004. The Safety Analysis for Byron includes a Spent Fuel Cask Drop event (UFSAR Section 15.7.5). The unrestrained stack-up configuration does not change this conclusion. Because the proposed configuration could tip-over in a spent fuel storage area independent of the FHB crane influence, and potentially affect the fuel pool and its ability to maintain water above the top of the spent fuel, it must be evaluated consistent with Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis in order to provide a measure of "defense-in-depth". The results of that investigation indicate that, in the unlikely event the unrestrained stack were to tip-over (due to operator error for example), the physical separation between the closest surface of the spent fuel pool wall and the unrestrained stack would preclude any direct impact. Consequently, in the very unlikely event the unrestrained stack were to tip over, it would not compromise the integrity of the pool sufficiently to uncover the fuel.

A dynamic nonlinear time history analysis was performed on the unrestrained stack-up configuration during a design basis seismic event to evaluate the rocking and sliding stability of the unrestrained stack-up configuration. The results of the analysis indicate that the maximum expected rocking angles are less than the minimum acceptable rocking angle with a Factor of Safety of at least 2.0 and maximum expected sliding distances are less than the minimum clearance distance to the FHB structure and/or adjacent FHB SSCs with a Factor of Safety of at least 3.0. Evaluations for DCS SSCs required to implement unrestrained stack-up configuration and the FHB structure determined that they remain within all Code allowables for the design basis loads during a seismic event. As a result the proposed activity will not impact plant operations, nor does it adversely affect the function of any plant equipment or structure that is used in establishing the Plant Design Basis.

The process of transferring the MPC from the HI-TRAC to the HI-STORM is a continuous process (without delay) and the MPC is continually under control of the FHB crane operator. A seismic event would be the most apparent event to challenge the

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Unit 0

Activity/Document Number: DRP 15-067 / EC 389969 / BFP FH-65 / BFP FH-66 / BFP FH-69 / BFP FH-81 / BFP FH-83
Revision Number: 0 / 0 / 12 / 0 / 12 / 3 / 4

Title: Unrestrained HI-TRAC/HI-STORM stack-up configuration in the Fuel Handling Building

integrity of the un-restrained stack. Because of the relatively short duration involved with the lift and set of the MPC into the HI-STORM (~2 hours), the infrequent number of dry cask loading activities per year (3), and the relatively infrequent occurrence of an SSE (1×10^{-5}/year), the probability of a seismic event occurring coincident with a dry cask loading campaign is 6.8×10^{-9}/year – which is below the threshold for concern. Therefore, the frequency of occurrence of a cask drop has not increased by the proposed activity. If a seismic event was to occur during the lift and set, an induced angularity increases the resultant load in the MPC downloader slings and the reeving wire ropes by an acceptable amount.

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 informal screening (not included with this package) concluded that the proposed activity resulted in an adverse change to an UFSAR described design function; could adversely affect how design functions are performed; and results in a change to the method of evaluation and thus warrants a 50.59 evaluation.

This 50.59 evaluation has determined that the unrestrained freestanding stack-up configuration of the HI-TRAC transfer cask (HI-TRAC) and HI-STORM 100 Version B overpack (HI-STORM) in the Fuel Handling Building (FHB) during the transfer of the Multi-Purpose Canister (MPC) (loaded with spent fuel assemblies) from the HI-TRAC to the HI-STORM can be implemented per plant procedures without obtaining a License Amendment and is based upon the following arguments:

- The proposed activity (removal of seismic lateral restraints on stack-up) does not increase the probability of occurrence of the spent fuel cask drop accident (or reasonable facsimile). The infrequent performance of the activity (6 hours/year) coupled with the probability of an SSE per year (1×10^{-5}/year) yields a very low probability (6.8×10^{-9}/year) of a seismically induced heavy load drop.
- There is no increase in the likelihood of a malfunction of SSCs important to safety. The only appropriate malfunction is failure of the return line to the spent fuel pool downstream of the two spent fuel pool heat exchangers. The probability of such an event would increase by much less than 10% by the proposed activity.
- The proposed activity is physically separate and independent of the fuel pool and fuel handling activities and therefore cannot influence the UFSAR described fuel handling accident or its consequences.
- The proposed activity can have a low probability of influence on SFP cooling malfunctions described in the UFSAR. However, the consequences (due to a loss of fuel pool cooling) of such a malfunction would be no different with the proposed activity because the malfunction is no different.
- The unrestrained freestanding stack-up configuration does introduce the possibility of a new accident, but the effects of that accident in relation to fuel pool integrity and the ability to maintain water above the fuel is not different from the previously evaluated accident (cask drop).
- The proposed activity does not introduce the possibility of a malfunction of an SSC important to safety with a different result than described in the UFSAR.
- The unrestrained freestanding stack-up configuration will not result in a DBLFPB as described in the UFSAR being exceeded or altered.
- The proposed methodology is not documented in an NRC safety evaluation, industry technical standard, or other docketed document, but sufficient NRC oversight and associated correspondence would indicate it is "approved by the NRC for the intended application".

Attachments: Attach all 50.59 Review forms completed, as appropriate.
Forms Attached: (Check all that apply.)

Tracking No. 6G-14-001

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Unit 0

Activity/Document Number: DRP 15-067 / EC 389969 / BFP FH-65 / BFP FH-66 / BFP FH-69 / BFP FH-81 / BFP FH-83
Revision Number: 0 / 0 / 12 / 0 / 12 / 3 / 4

Title: Unrestrained HI-TRAC/HI-STORM stack-up configuration in the Fuel Handling Building

- ☐ **Applicability Review**
- ☐ **50.59 Screening** — **50.59 Screening No.** ______ **Rev.** ______
- ☒ **50.59 Evaluation** — **50.59 Evaluation No.** 6G-14-001 **Rev.** 0

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: **DRP 15-107, EC 397549, EC 399325, EC 399326, EC 396144** **Revision Number:** **0, 0, 0, 0, 0**

Title: Implementation of WCAP-15063-P-A, Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," and WCAP-12610-P-A & CENPD-404-P-A Addendum 2-A, "Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO"

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

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

The activity is the implementation of the Nuclear Regulatory Commission (NRC) approved Westinghouse Improved Performance Analysis and Design Model (PAD 4.0) as described in WCAP-15063-P-A, Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," July, 2000; implementation of an updated NRC approved Integral Form ZIRLO Cladding Corrosion Model as described in Westinghouse WCAP-12610-P-A & CENPD-404-P-A Addendum 2-A, "Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO," October 2013; and the accompanying UFSAR changes to sections 1.6 and 4.2.

The Westinghouse PAD model is a best estimate fuel rod performance model used for both fuel rod performance analysis and safety analysis input. The PAD code consists of several fuel rod performance models integrated to predict fuel temperature, rod pressure, fission gas release, cladding elastic and plastic behavior, cladding growth, cladding corrosion, fuel densification, and fuel swelling as a function of linear power and time. PAD 4.0 introduces a new creep model to be used in the overall PAD fuel rod performance model. The new creep model accounts for advances in the understanding of in-reactor creep and represents a description of in-reactor creep relative to the information and data that is available.

WCAP-15063-P-A, Revision 1, with Errata describes PAD 4.0 and is intended for use in Westinghouse fuel design methodologies for Pressurized Water Reactors (PWRs). The results from the improved PAD 4.0 model are more consistent with in-reactor experience using a mechanistic approach. The NRC concluded that the use of PAD 4.0 is acceptable for fuel licensing applications up to rod average burnup of 62,000 MWD/MTU. The fuel rod average burnup for all Braidwood and Byron units is limited to 60,000 MWD/MTU per NF-AP-100-7000, "Westinghouse NSSS Reload Design Control Implementation." PAD 4.0 has been reviewed by the NRC and has been approved for use in Westinghouse fuel rod design analyses on a forward fit basis within the limitations and conditions described in the Safety Evaluation Report (SER).

The new Integral Form ZIRLO® and Optimized ZIRLO™ High Performance Fuel Cladding Material Cladding Corrosion Model was developed based on an improvement in the concept of the modified fuel duty index and has incorporated all the ZIRLO and Optimized ZIRLO cladding corrosion measured oxide thickness data. The new model more accurately reflects the temperature profile in the boiling region required to predict the measured oxides. The new model replaces the cladding corrosion model as described in WCAP-15063-P-A, Revision 1, with Errata.

WCAP-12610-P-A & CENPD-404-P-A Addendum 2-A presents the new corrosion model, and is intended for use in Westinghouse fuel design methodologies for Pressurized Water Reactors (PWRs). The cladding corrosion criterion adopted with the new Westinghouse cladding corrosion model ensures that cladding mechanical properties capable of retaining cladding integrity under operational conditions are maintained. The integral form ZIRLO and Optimized ZIRLO cladding corrosion model has been reviewed by the NRC and has been approved for use in Westinghouse fuel rod design analyses on a forward fit basis within the limitations and conditions described in the Safety Evaluation Report (SER).

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

The current UFSAR sections 1.6 and 4.2 include a reference to PAD 3.3/3.4. The revision to PAD 4.0 makes several changes to the model. The results from the improved PAD 4.0 model are more consistent with in-reactor experience using a mechanistic approach. The model changes are to the cladding creep, cladding irradiation growth, Zr-4 and ZIRLO clad thermal conductivity, Zr-oxide thermal conductivity, Equation Of State (EOS) gas pressure, the oxide-metal

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: DRP 15-107, EC 397549, EC 399325, EC 399326, EC 396144 **Revision Number:** 0, 0, 0, 0, 0

Title: Implementation of WCAP-15063-P-A, Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," and WCAP-12610-P-A & CENPD-404-P-A Addendum 2-A, "Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO"

ratio, and Zr-4 clad gas absorption models. While the form of the gap conductance and Fission Gas Release (FGR) models have not changed, the coefficients and uncertainties for these models have changed.

The current ZIRLO corrosion model, as described in UFSAR section 4.2, is based on a model that was originally developed for zircaloy-4 cladding. As utilities moved to increased fuel thermal duty associated with higher peaking factor, uprated core power, and longer cycle length, cladding corrosion has become one of the important factors in assessing the potential to achieve these goals. The rate of corrosion could eventually determine the fuel rod lifetime. The Westinghouse corrosion model was developed to predict best-estimate values for the observed data of ZIRLO® and Optimized ZIRLO cladding over a large range of operating conditions. The UFSAR revisions to sections 1.6 and 4.2 are required to ensure the correct PAD version and corrosion model are described.

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

The methods describing the PAD version and corrosion modeling in the UFSAR have been revised or replaced as outlined in WCAP-15063-P-A, Revision 1, with Errata and WCAP-12610-P-A and CENPD-404-P-A Addendum 2-A.

The Westinghouse PAD model is a best estimate fuel rod performance model used for both fuel rod performance analysis and safety analysis input. The PAD code consists of several fuel rod performance models integrated to predict fuel temperature, rod pressure, fission gas release, cladding elastic and plastic behavior, cladding growth, cladding corrosion, fuel densification, and fuel swelling as a function of linear power and time. The results from the improved PAD 4.0 model are more consistent with in-reactor experience using a mechanistic approach. Therefore, there is no impact as a result of implementing PAD 4.0.

The corrosion models are used to account for the fuel rod clad oxidation and hydriding design criteria for ZIRLO cladding. There is no impact to plant operations as a result of implementing the new corrosion model. The new corrosion model results documented in Westinghouse calculation notes CN-CC19-005 (Braidwood Unit 1) and CN-CB19-029 (Byron Unit 2) show that clad corrosion design limits for the associated units are met. This check is completed for every reload and due to the nature of the core design it is expected that all Byron and Braidwood units will have similar margin.

Based on a review of the UFSAR and input from NF-CB-14-109, the Byron and Braidwood UFSAR sections 1.6 and 4.2 will be updated to incorporate PAD 4.0 and the new corrosion model. The UFSAR section 15.6 reference to WCAP-12610 deals with steady-state ZIRLO cladding oxidation that is not impacted by the revised corrosion model. No safety analyses (Chapter 15) were required to be re-performed as a result of implementing PAD 4.0 and the new corrosion model. This change is not outside the bounds of WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," therefore the limits and analytical methods discussed within Technical Specifications Section 5.6.5 are not impacted as a result of implementing PAD 4.0 and the new corrosion model in WCAP 15063-P-A, Revision 1, with Errata, and WCAP-12610-P-A and CENPD-404-P-A Addendum 2-A.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: DRP 15-107, EC 397549, EC 399325, EC 399326, EC 396144 **Revision Number:** 0, 0, 0, 0, 0

Title: Implementation of WCAP-15063-P-A, Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," and WCAP-12610-P-A & CENPD-404-P-A Addendum 2-A, "Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO"

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.)

PAD 4.0, described in WCAP-15063-P-A, Revision 1, with Errata, has been approved by the NRC. The results from the improved PAD 4.0 model are more consistent with in-reactor experience using a mechanistic approach. The NRC has accepted these changes based on the results having been evaluated to be conservative or essentially the same. The NRC concludes that the use of PAD 4.0 is acceptable for fuel licensing applications up to rod average burnup of 62,000 MWD/MTU. The fuel rod average burnup for all Braidwood and Byron units is limited to 60,000 MWD/MTU per NF-AP-100-7000. Utilization of PAD 4.0 does constitute a departure from a method of evaluation described in the UFSAR; however, since it is used within the limits and constraints of the SER, it does not require NRC approval prior to its use. UFSAR sections 1.6 and 4.2 need to be updated to reference this new corrosion model.

The corrosion model described in WCAP-12610-P-A and CENPD-404-P-A Addendum 2-A has been approved by the NRC. Both Byron and Braidwood units meet the limitations and conditions of section 5.0 of the NRC SER for implementation of the new corrosion model for ZIRLO and Optimized ZIRLO cladding. UFSAR section 4.2 needs to be updated to reference this new corrosion model. The new clad corrosion model does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses, and thus does not require NRC approval prior to its use.

The attached 50.59 Screening concludes that the proposed activity does not involve a change to an SSC that adversely affects an UFSAR described design function (Question 1), nor does it 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 in a manner that is inconsistent with the analyses or descriptions in the UFSAR (Question 4) or require a change to the Technical Specifications or Facility Operating License (Question 5). The proposed Activity does not involve a change to a procedure that adversely affects how UFSAR described design functions are performed or controlled (Question 2). UFSAR Sections 1.6 and 4.2 currently contain references to an outdated version of PAD, which is a code used to provide initial thermal conditions (fuel centerline and volume average temperatures) and rod pressures for the start of the LOCA analysis. Since the implementation of PAD 4.0 involves changing an evaluation method described in the UFSAR, and the implementation of the new clad corrosion model involves a change to an element of an UFSAR described methodology, both of these changes are considered adverse and require further review in a 10CFR50.59 Evaluation.

The attached 50.59 Evaluation concluded that the PAD 4.0 model and the new clad corrosion model have been approved by the NRC for use on a forward fit basis, within the constraints described in the associated SERs. Since the fuel rod average burnup for all Braidwood and Byron Units is limited to 60,000 MWD/MTU, WCAP-15063-P-A, Revision 1, with Errata is acceptable for implementation at Byron and Braidwood Stations. WCAP-12610-P-A and CENPD-404-P-A Addendum 2-A is acceptable for all Braidwood and Byron Units since the new clad corrosion model has been approved by the NRC for all Westinghouse NSSS plants that use ZIRLO for their cladding material with consideration of the conditions as described in section 5.0 of the SER. Additionally, Technical Specification changes are not required since the Byron and Braidwood Technical Specification section 5.6.5 does not contain the corrosion model related topical report. The PAD 4.0 and the new corrosion model topical reports do not define the technical basis for a Core Operating Limits Report (COLR) limit. No other instances of reference to the topical report occurred within the Byron and Braidwood Technical Specifications.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: DRP 15-107, EC 397549, EC 399325, EC 399326, EC 396144 **Revision Number:** 0, 0, 0, 0, 0

Title: Implementation of WCAP-15063-P-A, Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," and WCAP-12610-P-A & CENPD-404-P-A Addendum 2-A, "Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO"

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.**	**BRW-S-2014-119 / 6E-14-092**	**Rev.**	**0 / 0**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**BRW-E-2014-120 / 6G-14-003**	**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 1 and Unit 2

Activity/Document Number: EC 400682 (U2), EC 401062 (U1), DRP 16-013, DRP 16-015
Revision Number: 0, 0, 0, 0

Title: Units 1 & 2 Reactor Vessel Closure Head Penetration Ultra High Pressure (UHP) Cavitation Peening

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 will apply the Areva Ultra-High Pressure (UHP) cavitation peening process to the Reactor Vessel Closure Head (RVCH) penetrations (nozzles) and J-groove welds. The target areas of the RVCH nozzles to be peened include the OD, ID, and J-groove weld with exception to previously repaired RVCH nozzles i.e., RVCH nozzles with previous surface remediation. The application of peening will provide asset life extension through elimination of the degradation process due to Primary Water Stress Corrosion Cracking (PWSCC). PWSCC is attributed to sustained surface tensile stress (which occurs from welding and grinding during original fabrication). PWSCC is mitigated at each applicable RVCH nozzle by inducing a compressive stress layer on the surface of the penetration tubes and the J-groove welds via the collapse of vapor bubbles which generate high local pressures. For Unit 2, penetrations 6 and 68 have been previously repaired by weld overlay and, as such, peening will be limited to those areas that did not receive weld overlay (ID; portions of the OD). For Unit 1, penetrations 31, 43, 64, and 76 have been previously repaired by weld overlay.

DRP 16-013 (U2) and DRP 16-015 (U1) will update UFSAR Section 5.2.3.4.4 to describe the application of UHP cavitation peening to the applicable RVCH nozzles and J-groove welds.

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

During the past two decades, Stress Corrosion Cracking (SCC) has become the most prevalent phenomenon affecting nuclear plant availability and plant lifetime management. SCC can lead to increased costs for operation, maintenance, assessment, repair and replacement of boiling water reactor (BWR) and pressurized water reactor (PWR) components. Alloy 600 and 82/182 materials, which are widely used in PWR systems, are susceptible to PWSCC.

The RVCH penetrations are SB166 or 167 (Alloy 600) tubes with 82/182 weld filler material at the J-groove weld and, thus, susceptible to PWSCC. Byron Unit 1 and Byron Unit 2 are classified as highly-susceptible heads due to the discovery of surface indications (cracks) during inspections. Therefore, ECs 400682 and 401062 applies the Areva UHP cavitation peening process to induce beneficial compressive residual stresses in the surface of the applicable Byron Unit 1 and Unit 2 penetration tubes and J-groove welds to mitigate PWSCC at the RVCH penetrations for the purpose of surface stress improvement.

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

The effect of the activity is the applicable RVCH penetrations and J-groove welds will have improved surface stress through application of UHP cavitation peening. Thus, PWSCC is mitigated at each RVCH nozzle by inducing a compressive stress layer in the surface of the penetration tubes and the J-groove welds. The depth of the compressive stress at the nozzle OD and J-groove weld is ~1mm (.04") and at the nozzle ID is ~1/4mm (.01"). There is no impact to any RVCH component geometry.

There is no impact during normal plant operation. The RVCH penetration surface peening application will be performed during an outage with the RVCH removed and resting on the head stand.

The function of the Control Rod Drive Mechanism (CRDM) is not affected. The CRDMs will continue to insert or withdraw rod cluster control assemblies within the core to control average core temperature and to shut down the reactor when online. The function of the Rod Cluster Control Assembly (RCCA) thermal sleeve guides and thermal sleeves is not affected. The RCCA thermal sleeve guides and thermal sleeves will continue to ensure proper control rod alignment during reactor head installation. The thermal sleeves will continue to provide protection of the RCCA drive rods from head cooling spray cross

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 400682 (U2), EC 401062 (U1), DRP 16-013, DRP 16-015
Revision Number: 0, 0, 0, 0

Title: Units 1 & 2 Reactor Vessel Closure Head Penetration Ultra High Pressure (UHP) Cavitation Peening

flow. The function of the Reactor Vessel Pressure Boundary (RVPB) is to maintain leak tight integrity for the Reactor Coolant System (RCS) and will continue to maintain leak tight integrity for the RCS after the RVCH target surface stress improvement is complete i.e., post nozzle and J-groove weld UHP cavitation peening.

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 Screening, question 1, conservatively concluded that cavitation peening – if administered incorrectly - could adversely affect the pressure boundary integrity of peened surfaces such as the CRDM housing. The following is a summary of the eight Evaluation questions:

1. The parts that are to be peened are reactor coolant system pressure boundary parts. The design function of the RCS pressure boundary parts is to maintain leak tight integrity. The only accidents that could conceivably be affected by the proposed peening are a loss of coolant accident or CRDM missile caused by a crack in the peened parts. The proposed peening reduces the likelihood of initiation of PWSCC and, thus, reduces the likelihood of a crack causing a loss of coolant accident/CRDM missile. The peening has a negligible effect on the dimensions and geometry of the peened parts. There are no changes to component design functions, or their methods of design or evaluation e.g., ASME Code equations or allowable stresses. Therefore, because the affected components continue to meet applicable NRC requirements, as well as the applicable design, material, and construction standards, it is concluded that the application of UHP cavitation peening to the applicable RVCH penetrations does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

2. The only conceivable malfunction of the reactor coolant system that could be affected by peening is an ATWS and the occurrence of cracking that initiates at the surface of the wetted parts. Rod control testing (1/2BOSR 1.4.3-1a) before plant start-up will provide assurance that the peening process did not adversely influence the rod drop time or the ability of the rods to insert. The performance of peening is expected to reduce the likelihood of cracking. There are no changes to component design functions, or their methods of design or evaluation e.g., ASME Code equations or allowable stresses. Compliance with the ASME Code is not affected by the peening. Therefore, the application of UHP cavitation peening to the applicable RVCH penetrations does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of the reactor coolant system pressure boundary that is important to safety as previously evaluated in the UFSAR.

3. The proposed peening has no effect on the consequences of any accidents. This is because the proposed peening only affects a thin layer i.e., ~1/4mm to 1mm (.01" - .04") at the surface of the peened parts and expected to have no effect on the performance of these parts during an accident and, thus, to have no effect on the consequences of the accident. Therefore, the application of UHP cavitation peening to the applicable RVCH penetrations does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

4. The appropriate SSC malfunctions described in the UFSAR are an ATWS, and the CRDM missile. If peening were to result in a failure of the CRDM housing or its interface with the reactor vessel head, it would result in reactor coolant losses far less than postulated from a break of the RCS cold leg (for example). Since the fuel integrity or containment integrity are not affected by the proposed activity, the consequences from the proposed activity cannot increase. Also, since the peening process has no effect on the ability to trip the turbine or initiate AF, ATWS consequences are unaffected.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 400682 (U2), EC 401062 (U1), DRP 16-013, DRP 16-015
Revision Number: 0, 0, 0, 0

Title: Units 1 & 2 Reactor Vessel Closure Head Penetration Ultra High Pressure (UHP) Cavitation Peening

5. The proposed peening only affects residual stresses in a shallow layer at the surface of the peened parts and, thus, cannot have any effect on accidents. There are no known cases where the proposed types of peening have caused different types of accidents to occur or different types of flaws to develop. Existing evaluations of potential loss of coolant accidents remain bounding. Therefore, the application of UHP cavitation peening to the applicable RVCH penetrations does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

6. The proposed peening only affects residual stresses in a shallow layer at the surface of the peened parts. The only effect of the presence of these stresses on performance of an SCC important to safety is a reduction in the likelihood of cracking. Existing evaluations of potential reactor coolant system malfunctions (including loss of coolant accidents) remain bounding. Therefore, the application of UHP cavitation peening to the applicable RVCH penetrations does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR.

7. The parts that are to be peened are reactor coolant system pressure boundary parts. The design function of the RCS pressure boundary parts is to maintain leak tight integrity i.e., serve as a fission product barrier. The proposed peening reduces the likelihood of initiation of PWSCC and, thus, reduces the likelihood that a crack may adversely affect the performance of the reactor coolant system as a fission product barrier. The peening has a negligible effect on the dimensions and geometry of the peened parts. There are no changes to component design functions, or their methods of design or evaluation e.g., ASME Code equations or allowable stresses. The reactor coolant system design pressure is not changed, and the pressure retaining capability of the reactor coolant system is not affected. Therefore, it is concluded that the application of UHP cavitation peening to the applicable RVCH penetrations does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

8. The proposed peening has no effect on the methods of evaluation described in the UFSAR that are used to establish design bases or perform accident analyses. There are no changes to component design functions, or their methods of design or evaluation e.g., ASME Code equations or allowable stresses. Compliance with the ASME Code is not affected by the peening. While the development of compressive stresses at wetted surfaces could affect the calculated fatigue usage factor, e.g., at reactor vessel nozzle dissimilar metal welds, the presence of the compressive stresses would clearly reduce the risks of fatigue cracking and, thus, not require a change in the method of evaluation. Therefore, it is concluded that the application of UHP cavitation peening to the applicable RVCH penetrations does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design basis or in the safety analyses.

Based on the above, the proposed activity may be implemented without prior NRC approval.

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

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	6E-15-063	**Rev.**	0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-15-002	**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 Units 1 and 2

Activity/Document Number: EC 401137 and 401138 **Revision Number:** 000/000

Title: Implement use of Westinghouse SHIELD for use in Reactor Coolant Pump Seal Configurations

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, to be implemented via EC 401137 (Byron Unit 1) and EC 401138 (Byron Unit 2), will replace the reactor coolant pump (RCP) Number 1 Seal Insert with a modified design called the SHIELD® Shutdown Seal (SDS). The SHIELD® SDS is referred to hereafter as the SDS. The SDS is a thermally activated, passive device located between the No. 1 and No. 2 pump seals, just upstream of the No. 1 leakoff line. The SDS is composed of a passive retractable spacer and a series of stacked rings comprising a wave spring, piston ring, polymer ring, and retaining ring. A thermal actuator holds the piston ring "open," permitting No. 1 seal leakoff to flow up the shaft to the No. 1 seal leakoff line during normal operation to maintain the existing seal leakoff flow characteristics. When actuated, the SDS is designed to be completely contained within the space once taken by the No. 1 seal insert prior to modification and will limit leakage through the reactor coolant pump seals by obstructing the annular flow path between the rotor and the No. 1 seal insert during loss of seal cooling scenarios. As long as seal cooling is maintained such that seal leak-off temperatures are kept within the allowable range for RCP operation, the SDS will not actuate and the overall reactor coolant system (RCS) inventory loss through the RCP seal leak-off will remain in the plant's normal operating range. Only in the case for which all seal cooling is lost and seal leakoff temperatures exceed the allowable operating range will actuation of the SDS occur and prevent excessive RCS inventory losses through the SDS seal package. The SDS is a defense-in-depth measure and its actuation is not credited for any licensing basis events.

The installed RCPs are Westinghouse Model 93A. The sealing surface for the SDS in the Model 93A RCPs is the RCP sleeve. The RCP shaft will be furnished with a new sleeve whose diameter and surface finish will be precisely controlled.

The number 1 seal is a film-riding face seal located above the lower RCP radial bearing. The film is produced by the system pressure drop across the seal and does not require seal rotation to establish the sealing function. To maintain the film, a controlled leakoff flow passes between the radially tapered seal faces. The SDS integrates new features into the number 1 seal insert and is located downstream of the film-riding face seal. The SDS design includes a shoulder machined into the inner diameter at the top flange and a bore machined into the groove diameter above the shoulder. SDS sealing rings and a thermal actuator are placed into this shoulder and bore, respectively.

The SDS, designed to actuate only when exposed to an elevated fluid temperature downstream of the RCP number 1 seal, deploys via retraction of a thermal actuator, which causes the SDS seal ring to constrict around the number 1 seal sleeve. SDS deployment controls shaft seal leakage and limits the loss of reactor coolant via the RCP seal package. SDS actuation occurs within a temperature range expected as a result of the coincident loss of all thermal barrier heat exchanger cooling and number 1 seal injection cooling. In its installed and non-activated state, the SDS resides completely out of the normal seal injection and shaft seal leakage flow paths. When activated as designed, on a stationary (or slowly rotating) shaft, the SDS limits RCP shaft leakoff to less than 1 gallon per minute (gpm) per pump.

This 50.59 review is applicable for the installation of a SDS on a single RCP or installation of an SDS on multiple RCPs.

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

The SDS permits plants to respond to a wide range of events involving loss of all seal cooling with only a diesel-driven auxiliary feedwater pump available. These events can include station blackout (SBO), fires that disrupt water supplies, loss of the Component Cooling System and loss of the Essential Service Water System. Since there are negligible RCS inventory losses through the RCP seals with the SDS actuated, RCS makeup is a lesser priority to achieve a stable state.

The purpose of this modification is to replace the RCP 1(2)RC01PA/B/D/C number 1 seal insert with a modified design that includes the SDS.

The SDS is being installed so that its low leakage features can be credited in subsequent evaluations for various applications such as Diverse and Flexible Coping Strategies (FLEX), Station Blackout (SBO), Fire Protection, and Probabilistic Risk Assessment (PRA) models.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 401137 and 401138 **Revision Number:** 000/000

Title: Implement use of Westinghouse SHIELD for use in Reactor Coolant Pump Seal Configurations

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

Normal Operation and Actuation of the SDS:

In its installed and non-activated state, the SDS resides completely out of the normal seal injection and shaft seal leakage flow paths. Until activated, the SDS maintains the same dimensions as the existing number 1 seal. The existing flow annulus between the number 1 seal insert and the PRC impeller shaft remains unaltered. The leakoff flow from the number 1 seal is not affected during normal operation of the RCP.

As long as seal cooling is maintained via chemical and volume control system (CVCS) seal injection or component cooling system (CC) thermal barrier cooling such that seal leak-off temperatures are kept within the allowable range for RCP operation, the SDS will not actuate and the overall RCS inventory loss through the RCP seal leak-off will remain in the plant's normal operating range. Only in the case for which all seal cooling is lost and seal leakoff temperatures exceed the allowable operating range will actuation of the SDS occur and prevent excessive RCS inventory losses through the seal package.

Upon a loss of all RCP seal cooling (CVCS seal injection and CC System thermal barrier cooling), the temperature of the fluid passing through the number 1 seal and past the SDS will begin to rapidly increase as hot reactor coolant begins to push the cold water in the pump annulus (between the pump impeller and the seal package) through the seal package. When the water temperature passing the SDS reaches a temperature range of 260 – 320 °F, the SDS actuates and stops virtually all flow through the number 1 pump seal, thus significantly reducing further RCS inventory losses. Actuation of the SDS occurs when thermal wax internal to the actuator expands to cause the actuator to retract and pull the spacer from between the butt ends of the piston ring, initiating deployment of the rings.

The SDS will permits plants to respond to a wide range of events involving loss of all seal cooling with only a diesel-driven auxiliary feedwater pump available. These events can include station blackout (SBO), fires that disrupt water supplies, loss of the Component Cooling System and loss of the Essential Service Water System. Since there are negligible RCS inventory losses through the RCP seals with the SDS actuated, RCS makeup is a lesser priority to achieve a stable state. However, this activity only addresses the installation of the SDS, and the SDS will not credited for any design basis event by this activity. Limitations and conditions for crediting the SDS for limiting RCS inventory losses through the No. 1 pump seal will be addressed by separate activities.

Inadvertent Actuation of the SDS:

Premature or unintended actuation of a single SDS upon the rotating RCP shaft could generate friction wear materials (from degradation of the stainless steel piston ring and the polymer sealing ring). In the event that the SDS were to malfunction and inadvertently actuate, the No. 1 seal requirement for "Essentially zero" leakage to containment will be maintained. The SDS underwent inadvertent actuation tests. No test showed evidence of damage to the number 1 seal (Reference TR-FSE-14-1-P). The number 1 seal leakoff flow is expected to stabilize at the value that existed prior to inadvertent actuation.

An inadvertent actuation will not have an adverse impact on the number 2 or 3 seals. As indicated above, frictional wear of the polymer and metal components of the SDS against the rotating shaft sleeve can generate small particles and threads of polymer material. Such wear products will mix with the primary coolant in the region between the RCP number 1 and 2 seals. The suspended particles/threads will be limited in size to those able to escape the radial clearance between the SDS retaining ring and the shaft sleeve. Since the leakoff flow past the number 2 seal is relatively low compared to that of the number 1 seal, the suspended particles/threads will be flushed out the number 1 seal leakoff line and not affect the operation of the number 1, 2 or 3 seals.

As summarized in the accompanying 50.59 Screening 6E-15-098, supporting analysis of the downstream effects of inadvertent actuation of all SDS units concluded that the passing of debris through components in the seal water return path would not adversely impact the component's UFSAR described seal water leak path design function with the exception of the Seal Water Return Filter (CV02F) and the No. 1 Seal Leakoff Return Flow Elements (FE 158-161) which are evaluated in the accompanying 50.59 Evaluation 6G-15-004.

Tracking No. 6G-15-004

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 401137 and 401138 **Revision Number:** 000/000

Title: Implement use of Westinghouse SHIELD for use in Reactor Coolant Pump Seal Configurations

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

50.59 Screening 6E-15-098 Summary/Conclusion:

Q1. Does the proposed Activity involve a change to an SSC that adversely affects an UFSAR described design function? A YES response was provided requiring further assessment in Evaluation 6G-15-004.

Screening 6E-15-098 segregated the response to this question into the following distinct UFSAR-described design functions:

1) **Loss of Reactor Coolant Accident Boundary:** It was concluded that until activated the SDS maintains the same dimensions as the existing No. 1 seal and when activated the SDS limits leakage through the RCP seals by obstructing the flow path between the RCP shaft and the No. 1 seal. For these reasons it was concluded that this activity does not adversely impact the UFSAR-described function of the No. 1 seal to function as a LOCA boundary.

2) **Seismic Integrity:** It was documented that the SDS is designed to remain functional following a seismic event. Seismic testing of the SDS was performed utilizing accelerations that bound the Byron seismic spectra. In addition, the weight of the SDS was documented to be insignificant when compared to the RCP/Motor assembly and installation of the SDS would not impact the seismic integrity of the RCP/Motor assembly. Therefore, it was concluded that the seismic integrity of the RCP and No. 1 seal as described in the UFSAR was not adversely impacted.

3) **Core Cooling – Adequate Flow and Natural Circulation:** ***Prior to actuation:*** it was documented that the SDS has no impact on the No. 1 seal operation or flow and since the SDS does not contact the RCP impeller shaft sleeve until it is activated Core Cooling and Natural Circulation is not affected. The plant operators will continue to utilize operating deign/limitations and seal monitoring instrumentation to monitor for seal damage. ***When actuated as designed***: The SDS limits leakage through the RCP seals. It was concluded that although the SDS piston ring snaps closed against the RCP shaft, as indicated in the UFSAR the RCP motor has sufficient power to continue pump operation even after a seizure of a pump seal. Therefore, when maintaining its intended design actuation function, this activity will not adversely impact UFSAR described Core Cooling and Natural Circulation design functions. ***Premature or Unintended actuation:*** It was concluded that wear materials could be generated that could become lodged in the No. 1 seal leakoff flow orifice potentially resulting in an erroneous high No. 1 seal leakoff alarm. Operator actions resulting from this alarm include tripping the reactor and shutting down the affected RCP(s). Since shutting down an RCP will adversely impact the available core cooling, it was concluded that this activity could adversely impact the UFSAR describe design function to provide core cooling, requiring further assessment in Evaluation 6G-15-004.

4) **Missile Generation:** It was concluded that the installation of the SDS would not impact the structural integrity of the RCP casing or heavy stator and since the SDS itself will not generate any new missiles, missile generation inside containment as described in the UFSAR was not adversely impacted.

5) **Limit Leakoff In a Controlled Fashion:** It was concluded that when maintaining its intended design configuration and actuation function, the SDS does not impact normal RCP seal leakoff flow. Although inadvertent actuation of the SDS could generate small debris particles, these particles will me mixed with the primary coolant and will be flushed out of the No. 1 seal leakoff line and will not affect the operation of the No. 1, 2 or 3 seals. It was concluded that implementation of the activity will not adversely impact the UFSAR described RCP seal function to limit leakoff in a controlled fashion.

6) **Seal Water Injection Flowpath (Reactivity/Inventory Control):** It was documented that there are diverse paths for reactivity (boration) and inventory control via seal injection and normal CVCS charging flowpaths. It was concluded prior to activation, installation of the SDS will not impact reactivity or inventory injection paths. Additionally, after actuation of the SDS the alternate CVCS injection flowpath still remains available. If the SDS

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 401137 and 401138 **Revision Number:** 000/000

Title: Implement use of Westinghouse SHIELD for use in Reactor Coolant Pump Seal Configurations

were to inadvertently actuate, the debris generated would be captured in the dead legs of the seal water return piping and the seal water return filter and debris particles would not be transported to the seal water supply filter. It was, therefore, concluded that implementation of this activity does not adversely impact the UFSAR described seal water injection function for reactivity/inventory control.

7) **Seal Water Leakoff (or "Return/Filtration") Flowpath:** It was concluded that prior to activation, the SDS does not impact seal leakoff flow rate or instrumentation operation. An assessment of inadvertent activation and associated debris generation on the seal water leakoff flowpath was summarized. It was concluded that the potential debris generated would not impact any SSCs UFSAR described design function with the exception of the Seal Water Return Filter 1(2)CV02F and the Seal Water Leakoff Return Flow Elements 1(2)FE-158-161. It was concluded that plant operator actions in response to a fouled seal water return filter would be no different than any other cause of filter fouling, therefore, the UFSAR described filter function was not adversely impacted by this activity. However, debris blockage in the seal water return flow elements could produce an erroneous high No. 1 seal leakoff flow alarm resulting in tripping of the reactor and tripping of the affected RCP. Therefore, it was concluded that the UFSAR described function of the seal water return path instrumentation function to provide the plant operator with indication and alarm to assess for seal damage was adversely impacted and would require further assessment in Evaluation 6G-15-004.
8) **Radiation Sources:** It was concluded that even if the SDS was to inadvertently actuate, the seal water return filter would not be fouled with radioactive materials exceeding that assumed in the current dose analysis . Therefore, this activity will not adversely impact the UFSAR defined radiation sources.

Q2. Does the proposed Activity involve a change to a procedure that adversely affects how UFSAR described SSC design functions are performed or controlled? A NO response was provided.

It was concluded that the SDS does not affect the manner in which the RCPs are operated or controlled. Although operator actions (including the potential for plant trip and RCP trip) would be performed in response to indications of a RCP seal failure following SDS inadvertent actuation, these actions would be performed in accordance with existing plant procedures regardless of the cause of the seal failure. Therefore, it was concluded that implementation of this activity does not involve a change to a procedure that adversely affects how UFSAR described SSC design functions are performed or controlled.

Q3. Does the proposed Activity involve a adverse change to an element of a UFSAR described evaluation methodology, or use an alternative evaluation methodology, that is used in establishing the design bases or used in the safety analyses? A NO response was provided.

It was concluded that the methods of evaluation of the RCP seal design functions are not described in the UFSAR, therefore, the methods utilized to assess the installation of the SDS do not result in a departure in a method of evaluation described in the UFSAR.

Q4. Does the proposed Activity 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? A NO response was provided.

It was concluded that the proposed activity does not represent a test or experiment; it is a permanent change to the facility. The RCP seals will not be utilized or controlled in a manner that is outside the reference bounds of their design and will not be inconsistent with analyses or descriptions in the UFSAR. The SDS is utilized within the bounds of the existing RCP seal design and is consistent with the RCP seal description in the UFSAR.

Q5. Does the proposed Activity require a change to the Technical Specifications or Facility Operating License? A NO response was provided.

It was concluded that implementation of the activity will not challenge existing TS 3.5.5 RCP seal injection flow limits and associated operator actions, completion times, or surveillance requirements specified for the RCP seals. RCP seal injection flow resistance as described in TS 3.5.5 will not be affected by the implementation of the activity, and the activity will not require changes to the plant Technical Specifications or Operating License.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 401137 and 401138 **Revision Number:** 000/000

Title: Implement use of Westinghouse SHIELD for use in Reactor Coolant Pump Seal Configurations

In summary, a YES responses was provided for 50.59 Screening question #1, which required further assessment in Evaluation 6G-15-004.

50.59 Evaluation 6G-15-004 Summary/Conclusion:

Q1. Does the proposed activity result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR? A NO response was provided.

The previous accidents assessed included: UFSAR Section 15.3.1 "Partial Loss of Forced Reactor Coolant Flow", UFSAR Section 15.3.2 "Complete Loss of Forced Reactor Coolant Flow", and UFSAR Section 15.2.6 "Loss of Nonemergency AC Power to Plant Auxiliaries". It was concluded that the installed SDS, should it actuate normally or inadvertently, would not cause an electrical fault, loss of off-site power or mechanical failure which are various initiating events for this accidents. In addition, although it was acknowledged that inadvertent failure of the SDS could provide a negligible amount of resistance against the rotating RCP shaft, this force is incapable of causing an RCP shaft seizure assessed in UFSAR Section15.3.3.

Q2. Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR? A NO response was provided.

It was stated that the SDS is designed and tested to actuate only after all seal cooling is lost and it is not expected to cause a SSC important to safety to malfunction. However, as concluded in the 50.59 Screening, inadvertent SDS activation on a rotating RCP shaft sleeve and the possibility of resulting debris to damage the RCP seals or supporting components were reviewed as potential paths to an SSC malfunction. Previous malfunctions related to the RCP evaluated in the UFSAR included a reactor trip caused by loss of AC power of mechanical failure and RCP rotor breaks or locks caused by seizure of the shaft/rotor. It was concluded that the SDS does not impact any electrical components, therefore, neither installation of the SDS nor inadvertent actuation of the SDS will impact the likelihood of occurrence of a malfunction of equipment which may lead to a loss of AC power. Although it was concluded in the 50.59 Screening that inadvertent actuation of the SDS (and resulting debris) could result in malfunction of the No. 1 seal leakoff flow instrumentation which could lead to a reactor trip and RCP trip, the evaluation concluded that based on the design and testing of the SDS, the probability for all required events to occur concurrently is extremely low and the likelihood of occurrence of failure of the described No. 1 seal flow element can be considered less than minimal.

Q3. Does the proposed activity result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR? A NO response was provided.

It was concluded that the SDS will have no adverse impact on core cooling capability of the RCPs and, thus, none of the UFSAR-described safety analysis consequences will be impacted. The SDS neither contains nor releases radioactive materials, and in the event of a breach of the pressure boundary, the SDS rings and actuator cannot contribute to the radiological consequences of these events.

Q4. Does the proposed activity result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR? A NO response was provided.

It was concluded that inadvertent actuation of the SDS may result in operator actions including reactor trip and RCP trip, however, the consequences are still bounded by previous accidents evaluated in the UFSAR.

Q5. Does the proposed activity create a possibility for an accident of a different type than any previously evaluated in the UFSAR? A NO response was provided.

It was concluded that the SDS is passive and inactive during normal plant operation. Whether or not the SDS actuates as designed, inadvertently actuates, or fails to actuate, the proposed activity does not create a new or different accident than any previously evaluated in the UFSAR.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 401137 and 401138 **Revision Number:** 000/000

Title: Implement use of Westinghouse SHIELD for use in Reactor Coolant Pump Seal Configurations

Q6. Does the proposed activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR? A NO response was provided.

Considering the unlikely inadvertent actuation of an SDS which results in a manual RCP trip due to a debris-related erroneous flow element function and associated high No. 1 seal leakoff alarm, the malfunction result is no different than previous RCP trips evaluated in the UFSAR. Additionally, the UFSAR indicates that there are no credible sources of shaft seizure other than impeller rubs. The SDS does not introduce a credible source for seizure nor can it cause a different or more severe result.

Q7. Does the proposed activity result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered? A NO response was provided.

It was concluded that the installation of the SDS, whether the SDS activates as designed, inadvertently activates or fails to activate, will not exceed or alter any fission product barrier.

Q8. Does the proposed activity result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses? A NO response was provided.

It was concluded that the methods of evaluation for the RCP seal design functions are not described in the UFSAR and, therefore, the activity to install the SDS does not result in a departure in a method of evaluation described in the UFSAR.

In summary, it was 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-15-098	**Rev.** 0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-15-004	**Rev.** 0

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

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50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: Unit 1: ECs 362146, 362147, 362148, 362149 **Revision Number:** 0, 0, 0, 0
Unit 2: ECs 362150, 362151, 362152, 362153 **Revision Number:** 0, 0, 0, 0
UFSAR Changes 16-019, 16-020 and 16-021, Technical Specification Bases Change 15-008, FDRP 27-014, 27-020 and 27-021

Title: Replacement of ESF Instrument Power (IP) Inverters and Connection to Constant Voltage Transformers (CVT)

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 Engineering Changes (EC's) listed above replace the existing Byron Unit 1 and Unit 2 Instrument Power (IP) system 7.5 KVA Westinghouse instrument inverters, 1(2)IP05E, 1(2)IP06E, 1(2)IP07E, and 1(2)IP08E, with new 10 KVA inverters manufactured by AMETEK (Solid State Controls, Inc.). The existing inverter supplemental cooling fans and air conditioning units will also be removed since the replacement inverters are supplied with two cooling fans each.

The ECs will also connect Unit 1 and Unit 2 IP system Constant Voltage Transformers (CVTs) directly to their associated instrument inverters as the normal bypass power source. This allows for automatic transfer of power from the inverter to the CVT if the inverter generated output were to fail. Manual selection to the CVT for supplying the bus power is still available at the inverter, via the bypass switch.

Replacement of the inverters, removal of the cooling units, and connections for the inverter/CVT instrument bus feeds will also require a change to the Technical Specification Basis, Updated Final Safety Analysis Report (UFSAR), and the Fire Protection Report (FPR). This change in Instrument Bus power feed configuration is identical to the configuration employed by Braidwood Nuclear Station as identified in the Byron/Braidwood UFSAR.

The affected inverters and CVTs are safety-related, seismically qualified, and located in the MEER on elevation 451' of the Auxiliary Building.

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

Replacement parts for the existing Westinghouse inverters are becoming difficult to obtain. Therefore, the decision has been made to replace the existing inverters with new equipment.

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

This change affects IP system CVTs, inverters and instrument buses. This equipment is part of the IP system and interfaces with the Annunciator (AN), Auxiliary Power (AP), Direct Current (DC), and Miscellaneous Electrical Equipment Room (MEER) HVAC (VE) systems. Design basis and licensing requirements for the IP system will be met by the new inverters, and no adverse impact on any of the interfacing systems or any other plant Systems, Structures, Components (SSCs) will result from the implementation of these ECs. The intent of these changes is to replace the existing IP inverters and to connect the output of the IP CVTs as the bypass power source input of the inverters. The replacement inverters will interface with their inputs (AC and DC sources) and outputs (IP bus, Annunciator) in the same manner as the existing inverter. Although the manufacturer, design, and configuration of the inverters will change, the instrument inverters will continue to perform the same design function(s).

Failure mechanisms (i.e., component failures) may have been altered due to the change in design, type, and configuration of the internal components of the replacement inverters. This proposed change introduces two new failure modes for the IP system: failure of the inverter static switch and failure of the single cable/breaker supplying power to the instrument bus distribution panel. In addition, the new configuration employs an automatic switchover to the CVT power supply instead of the current design which requires a manual switchover to the CVT. A manual bypass of the static switch is also available to bypass the static switch and

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: **Unit 1: ECs 362146, 362147, 362148, 362149** **Revision Number: 0, 0, 0, 0**
Unit 2: ECs 362150, 362151, 362152, 362153 **Revision Number: 0, 0, 0, 0**
UFSAR Changes 16-019, 16-020 and 16-021, Technical Specification Bases Change 15-008, FDRP 27-014, 27-020 and 27-021

Title: Replacement of ESF Instrument Power (IP) Inverters and Connection to Constant Voltage Transformers (CVT)

power the instrument bus if required. The end result (effect) of these new failure modes of the replacement inverters (e.g., loss of inverter function) is the same as with the existing inverters and is bounded by existing analyses. See 50.59 Evaluation 6G-15-005 for discussion of these new failure modes. Failure modes for electrical components (e.g., capacitors, diodes, relays, resistors, transformers, etc.) remain unchanged.

Currently, the outputs of the CVTs and inverters are connected to two separate input breakers at the 120V AC instrument buses. The instrument bus can be energized from either the inverter or the CVT. The two input breakers at the 120V AC instrument buses distribution panels are tied together so both power sources cannot be placed on the instrument bus at the same time. In order to transfer power from the inverter to the CVT, the instrument bus must be de-energized. Under the new configuration, with the CVT output connected as the inverter bypass input, the CVT output is synchronized with the inverter output and the bus feed may be transferred to either source without de-energizing the bus (bumpless transfer). Manual selection of the CVT power is still available through the inverter unit. If the inverter generated output is lost, the CVT can manually be placed on the bus as the AC feed via the bypass switch, which bypassed the output from the inverters static switch.

Although the configuration of the CVTs, inverters, and instrument buses will change, they will continue to provide a reliable power supply of power to the instrument bus as described in UFSAR Chapter 8.3.1 for instrument and control power. Failure mechanisms (i.e., component failures) have been altered due to the change in connection configuration. Failure of the transfer switch or cable between the inverter and the instrument bus will result in the inability to supply power to the bus from either source. The result of any new failure modes created will result in loss of instrument bus voltage, which has been previously reviewed in the UFSAR Chapter 7.2.2.4.2. See 50.59 Evaluation 6G-15-005 for discussion of the CVT to inverter connection, inverter static switch and the automatic vs manual aspect of this change.

These changes will require a change to the wording in UFSAR Chapter 8.3.1.1.2.3, UFSAR Table 8.3-5, and Technical Specification Bases B 3.8.7 to reflect inverter to CVT automatic transfer capability. Reference UFSAR Changes 16-019, 16-020 and 16-021 and Technical Specification Bases Change 15-008. However, the UFSAR described design functions of the IP system are not adversely affected or altered. This change in Instrument Bus power configuration is identical to the configuration employed by Braidwood Nuclear Station as identified in the Byron/Braidwood UFSAR.

The affected IP system equipment and interfacing equipment and systems will continue to perform their design functions as described in the UFSAR.

The supplemental air conditioning units presently installed for the inverters have flexible duct work that is described in the Fire Protection Report (FPR). Since the ductwork will be removed when the air conditioning units are removed, the FPR is being revised to remove the ductwork from the listing of combustibles. Reference FDRP Changes 27-014, 27-020 and 27-021.

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.)

<u>50.59 Screening Conclusion:</u>

The replacement inverters will meet all performance, and qualification requirements as the existing inverters in accordance with the applicable Regulatory Guidelines, Licensing documents, and Exelon and industry standards. This change is considered adverse and screens in for the following reasons:

1. The new configuration replaces a manual function with an automatic function.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: **Unit 1: ECs 362146, 362147, 362148, 362149** **Revision Number: 0, 0, 0, 0**
Unit 2: ECs 362150, 362151, 362152, 362153 **Revision Number: 0, 0, 0, 0**
UFSAR Changes 16-019, 16-020 and 16-021, Technical Specification Bases Change 15-008, FDRP 27-014, 27-020 and 27-021

Title: Replacement of ESF Instrument Power (IP) Inverters and Connection to Constant Voltage Transformers (CVT)

2. The new configuration also employs a single feed to the instrument bus via the inverter for both the CVT power supply and the normal inverter power supply compared to the current configuration which has the CVT feed and the inverter feed completely separate.
3. The new configuration employs a new device (Static Switch) which was not previously utilized. This could provide new failure modes which were not previously considered.

Therefore, 50.59 Evaluation 6G-15-005 has been completed for discussion of the CVT to inverter connection, inverter static switch and the automatic vs manual aspect of this change.

Existing procedures for aligning, operating, controlling, and maintaining the IP inverters will require revision due to the change in inverter configuration and operation. Nevertheless, revising plant procedures to reflect the new configuration of the new inverters, CVTs and IP Buses will NOT result in a change that would adversely affect how their design functions are performed or controlled. This change re-configures the IP buses to eliminate the delay caused by dispatching an Operator to switch the IP source power manually. This change in Instrument Bus power configuration is identical to the configuration employed by Braidwood Nuclear Station as identified in the Byron/Braidwood UFSAR.

The replacement inverters and connection of the CVTs to the inverters being implemented in these design changes do not revise or replace any evaluation methodologies in the UFSAR or use an alternative evaluation methodology. There is no impact on existing accident analysis or transient analysis. The inverters and CVT's were qualified using the same methodology as the existing equipment as described in UFSAR Chapter 3.10.

Technical Specifications 3.8.7, 3.8.8, 3.8.9 and 3.8.10 and the associated bases were reviewed. There is no change to the existing Technical Specification Actions or Surveillance requirements. Technical Specification Bases 3.8.7 – ACTIONS – A.1 will be revised to reflect inverter to CVT automatic transfer capability. These new inverters and the connection of the CVT output to the inverters does not affect SSCs in a manner that would require a change to any Technical Specification Limiting Condition for Operation. Operating License requirements for either unit are not affected.

50.59 Evaluation Conclusion:

1. Failure of the inverter static switch or cable to the instrument bus could result in the inability to supply power to the bus from either source. UFSAR Chapter 7.2.2.4 reviews a failure of an instrument inverter. In the current configuration, failure could lead to a plant transient or even a reactor trip, however with the automatic transfer via the Static Switch the CVT, the frequency of occurrence of the transient or reactor trip has been reduce. The new static switch is safety related and has been qualified to the same requirements as the original safety related components. The likelihood of a cable or breaker failure of the new configuration is considered negligible. Therefore, the replacement of the CVTs, instrument bus inverters and associated bus supply cable configuration, the automatic vs manual transfer to the CVT and the introduction of the two new failure modes does not introduce the possibility of a change in the frequency of an accident due to the failure of the inverter static switch or cable supplying power to the instrument bus.

2. This proposed change introduces two new failure modes for the IP system: failure of the inverter static switch and failure of the single cable supplying power to the instrument bus distribution panel. Failure of the inverter static switch or cable to the instrument bus could result in the inability to supply power to the bus from either source. However, the resultant loss of an instrument bus has already been reviewed as described in UFSAR Chapter 7.2.2.4. Therefore, the replacement of the CVTs and instrument bus inverters along with the associated bus supply cable configuration will not result in more than a minimal increase in the likelihood of occurrence of a malfunction.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: **Unit 1: ECs 362146, 362147, 362148, 362149** **Revision Number: 0, 0, 0, 0**
Unit 2: ECs 362150, 362151, 362152, 362153 **Revision Number: 0, 0, 0, 0**
UFSAR Changes 16-019, 16-020 and 16-021, Technical Specification Bases Change 15-008, FDRP 27-014, 27-020 and 27-021

Title: Replacement of ESF Instrument Power (IP) Inverters and Connection to Constant Voltage Transformers (CVT)

3. The replacement of the CVTs and instrument bus inverters along with the associated bus supply cable configuration does not introduce the possibility of a change in the consequences of an accident. This is due to the fact that the replacement CVTs and instrument bus inverters are designed to the same original design and licensing basis requirements for safety related electrical components as the original equipment. There will be no impact on accident mitigation systems that would affect any dose analysis. In addition, no fission product barriers will be affected.

4. The replacement of the CVTs and instrument bus inverters along with the associated bus supply cable configuration does not introduce the possibility of an increase in the consequences of a malfunction. The reliability of the new equipment is expected to be better than the existing equipment due to the use of solid state components which are expected to be more reliable than equipment which was designed and manufactured over 30 years ago.

5. The replacement of the CVTs and instrument bus inverters along with the associated bus supply cable configuration does not affect accident initiation sequences or response scenarios as modeled in the safety analyses. This automatic transfer will result in a more reliable power supply and eliminate the loss of power to the bus upon loss of inverter output Therefore, there is no increase in the possibility of an accident of a different type than is already analyzed in the UFSAR.

6. The new inverter and CVT employs a new automatic feature and a new component (Static Switch). The new configuration also employs a single feed for the normal and backup (CVT) power supply to the instrument bus instead of two separate feeds. Failure of the new static switch or the single feed results in a loss of the power supply to the associated instrument bus which has been previously evaluated in Chapter 7 of the UFSAR. Therefore, the replacement of the CVTs and instrument bus inverters, along with the associated bus supply cable configuration, does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

7. This proposed change does not result in a change that would cause any system parameter to change. Therefore, the inverter replacements do not result in a DBLFPB as described in the UFSAR being exceeded or altered.

8. The replacement of the instrument bus inverters and CVTs, along with the associated bus supply cable configuration, does not involve a method of evaluation as defined in the UFSAR. The methodology to qualify the new inverters and CVT's is consistent with the original qualification methods. Therefore, this 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 attached 50.59 screening and evaluation NRC notification or NRC approval is not required and the changes may be implemented per applicable governing procedures.

Attachments:

Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	**6E-15-052**	**Rev.**	**0**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-15-005**	**Rev.**	**0**

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

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron / Unit 2

Activity/Document Number: EC 402345/ DRP 16-062 **Revision Number:** 002

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

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

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

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

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

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

The UFSAR description of the methodology for determining the Technical Specification Allowable Values and nominal setpoints will be revised. Currently the methodology description in the UFSAR is not consistent with the NRC-approved setpoint methodology as described in the Technical Specification Basis. The UFSAR change in DRP 16-062 will correct the description in the UFSAR to ensure it is consistent with the description in the Technical Specification Basis.

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

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

During the review of the proposed relay replacement, the methodology description in the UFSAR for determining the Technical Specification Allowable Value and nominal setpoint was determined to be inconsistent with the NRC-approved setpoint methodology as described in the Technical Specification Basis.

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

Like the existing relays, the new underfrequency relays will receive the bus frequency input from the existing 6.9 kV bus potential transformers and, on a detected low frequency input into the RPS, will initiate a reactor and RCP trip. The trip setpoint and time delay will remain the same. The new relays will be powered from 125 VDC ESF distribution panels, whereas the existing relays are powered only from the 6.9 kV bus potential transformers.

The effect of the UFSAR changes described in DRP 16-062 is to provide consistency between the UFSAR description of the RPS setpoint methodology and the description of the NRC-approved setpoint methodology in the Technical Specification Bases.

There is no impact on the design bases or safety analyses described in the UFSAR.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron / Unit 2

Activity/Document Number: EC 402345/ DRP 16-062 **Revision Number:** 002

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

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

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

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

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

The T.S allowable value and the nominal setpoint for the new relays were calculated using an NRC-approved methodology which determined that no change was required. However, the methodology described in the UFSAR does not agree with this NRC-approved methodology. The screening determined that this constituted use of an alternative methodology; the 50.59 evaluation determined that using the NRC-approved methodology was acceptable, and the UFSAR is being modified to document this approved methodology.

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

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

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	**6E-15-168**	**Rev.**	**0**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-15-006**	**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 1 and 2

Activity/Document Number: TCCP EC 404997 **Revision Number:** 0

Title: Remove AF Diesel Air Intake Elbow And Blank Off TB Air Intake

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

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

Temporary Modification (TCCP) EC 404997 will temporarily modify the combustion air intake of the Unit 1 and Unit 2 diesel driven Auxiliary Feedwater (AF) pumps (1/2AF01PB) to allow air suction from the Auxiliary Building cubicle in which the pump/diesel engine is contained rather than from the Turbine Building. Therefore, in the event that the pumps are started, air from the Auxiliary Building will enter the diesel engine and will be exhausted through line 1/2DO85A-16" and released out to the atmosphere via a vent on the Auxiliary Building roof. The combustion air intake in the Turbine Building will also be modified to prevent communication of the Turbine Building with the AF pump room, by removing the expanded metal screen in the Turbine Building and installing a suitable blank-off plate. To allow a sufficient air supply for the diesel engine combustion process, the door from Auxiliary Building General Area to the AF pump cubicle will be opened, which necessitates disabling the CO_2 fire protection to the room. The TCCP will establish compensatory measures for Auxiliary Building 383' elevation general area and the AF diesel rooms in accordance with 0BOL 10.d and 0BOL 10.g for degraded fire barriers and CO_2 system unavailable. The fire watch will ensure timely identification and response to a general area fire to preclude the fire from affecting the AF diesels and other nearby equipment.

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

The 1/2AF01PB diesel driven pump combustion air intake (1/2DOB1A-14) takes suction from the Turbine Building. During the 2016 Braidwood NRC Component Design Basis Inspection (CDBI), the inspection team requested information supporting the acceptability of locating the combustion air intake for the B-train AF diesel engines in the Turbine Building. During the review of available documentation related to the B-train AF diesel engine air intake, it was identified that the documentation did not support operation of the diesel with High Energy Line Break (HELB) environmental conditions postulated in the Turbine Building. This issue is also applicable for Byron as the design is the same (IR 2636112).

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

The change will allow proper operation of the diesel driven AF pumps during normal and accident conditions. However, the proposed activity will result in air from the diesel driven AF pump cubicle being used for combustion air for the diesel engine to be exhausted directly to the outside, bypassing the Auxiliary Building Ventilation (VA) exhaust flowpaths, which are monitored effluent release pathways. The new release paths are un-monitored effluent release pathways, therefore, alternate monitoring capability is required to be implemented.

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 will maintain the design functional performance requirements assumed in the UFSAR for the diesel driven AF pumps. Although no physical change are made to the VA exhaust filtration SSC's, the proposed

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: TCCP EC 404997 **Revision Number:** 0

Title: Remove AF Diesel Air Intake Elbow And Blank Off TB Air Intake

activity changes proposed activity will affect the UFSAR described design features of VA exhaust system of precluding direct exfiltration of potentially contaminated air from the Auxiliary Building following an accident which could result in abnormally high airborne radiation in the Auxiliary Building. Although it was determined that the existence of these unfiltered release pathways being created did not impact the radiological dose consequences following a Loss of Coolant Accident (LOCA), this was determined to be an adverse impact requiring a 10 CFR 50.59 Evaluation.

The 10 CFR 50.59 Evaluation concluded that since the operation of the diesel driven AF pumps and the VA system were not degraded, the activity has no impact on the potential for or consequences malfunctions of SSC's important to safety. Also, since the AF and VA systems are not accident initiators, there is no increase on the likelihood of an accident previously evaluated in the UFSAR from occurring. As indicted above, since the proposed activity was evaluated to have no impact on the radiological dose calculations, it was determined to have no impact on the consequences of an accident previously evaluated in the UFSAR. As failure scenarios for the AF and VA systems are unaffected, the proposed activity does not create the potential for an accident of a different type than previously evaluated in the UFSAR. As the modified SSC's perform a passive function, no new malfunctions of SSC's important to safety with different results than any previously evaluated in the UFSAR were identified. The modified SSC's do not result in operation that can impact the containment, Reactor Coolant System, or fuel cladding, therefore, the proposed activity does not result in a fission product barrier being exceeded or altered. Finally, no methods of evaluation not described in the UFSAR were utilized to evaluate the proposed activity.

Based on this evaluation, the proposed activity requires a 10 CFR 50.59 evaluation, however, the resulting evaluation determined that the proposed activity can be performed without prior NRC permission per the applicable governing procedures.

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

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	6E-16-027	**Rev.**	0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-16-001	**Rev.**	0

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 400277/EC 399174/DRP 16-012 **Revision Number:** 000 / 000 /000

Title: Revising AST Accident Dose Calculations

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

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

This activity will implement Engineering Changes (EC) 400277 (BYR), EC 399174 (BRW) and UFSAR change DRP 16-012. The above ECs incorporate the four revised design analyses listed below. The analysis determines the radiological consequences using Alternative Source Term (AST) methodology. The potential accidents for which Control Room (CR), Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses have been recalculated for Byron and Braidwood are as follow:

a) BYR04-051/BRW-04-0038-M, Rev. 5, "Re-analysis of Loss Of Coolant Accident (LOCA) Using Alternative Source Terms".
b) BYR04-045 & BRW-04-0039-M, Rev. 4, "Re-analysis of Control Rod Ejection Accident (CREA) Using Alternative Source Terms".
c) BYR04-047/BRW-04-0041-M, Rev. 4, "Re-analysis of Fuel Handling Accident (FHA) Using Alternative Source Term".
d) BYR04-049/BRW-04-0043-M, Rev. 4, "Re-analysis of Locked Rotor Accident (LRA) Using Alternative Source Term".

The proposed activity also involves UFSAR change (DRP 16-012) to incorporate the revised doses.

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

RADTRAD is the computer code used for the Analysis Of Record (AOR) listed above. An error was discovered in the version of RADTRAD that was used in the AOR for the LOCA dose described in UFSAR 15.6.5 (IR 01320861). The correction to this error required reanalysis which resulted in use of the latest calculated Offsite Atmospheric Dispersion Factors (χ/Q) due to the commitment made to NRC (UFSAR 2.3.6.3). Based on this commitment, the χ/Q values needed to be re-evaluated based on the finer wind speed categories provided in the latest Regulatory Guidance (RG) 1.23, Revision 1, the next time calculations associated with the dose consequences for the LOCA, MSLB, CREA, LRA, SGTR and FHA were revised. In support of the License Amendment Request (LAR) for Measurement Uncertainty Recapture (MUR) Power Uprate, Exelon performed a re-evaluation of the offsite χ/Q values for the radiological dose analyses for the Main Steam Line Break (MSLB) and Steam Generator Tube Rupture (SGTR) accidents. Hence, the new Offsite χ/Q values needed to be incorporated in the accident dose analyses for the LOCA, CREA, LRA and FHA due to the commitment made to the NRC.

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

This activity does not involve any physical change, nor has any impact on plant operations, nor does it impact the initiation or progression of any accidents. It only changes the calculated dose following postulation of the accidents discussed above.

An error was discovered in the version of RADTRAD that has been used in the AOR described in UFSAR 15.6.5. The correction to this error resulted in the analyzed dose to be higher (conservative) than those

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 400277/EC 399174/DRP 16-012 **Revision Number:** 000 / 000 /000

Title: Revising AST Accident Dose Calculations

previously calculated. The correction in the RADTRAD code is to a design element. Per NEI 96-07 Revision 1, "In general, licensees can make changes to elements of a methodology without first obtaining a license amendment if the results are essentially the same as, or more conservative than, previous results." As shown in table below, the results are conservative and remain within the limits of 10 CFR 50.67(b)(2) and RG 1.183.

The resultant doses from the existing and the re-analyzed Accident Radiological Dose Analyses and the limits of 10 CFR 50.67(b)(2) and RG 1.183 are listed in the table below:

		Rem TEDE		
Accident	Dose	CR	EAB	LPZ
a. LOCA	Existing	4.78	12.20	2.99
	New	¥ 4.88	¥* 14.76	¥* 4.76
	Limit	5	25	25
b. CREA	Existing	4.538	4.647	1.983
	New	4.538	* 5.358	* 2.278
	Limit	5	6.3	6.3
c. FHA	Existing	4.28	4.24	0.356
	New	4.28	* 4.89	* 0.87
	Limit	5	6.3	6.3
d. LRA	Existing	2.79	1.456	0.525
	New	2.79	* 1.679	* 0.602
	Limit	5	2.5	2.5

NOTEs: "*" The dose increase is due to use of new Offsite Atmospheric Dispersion factor (χ/Q). In accordance with this commitment, the χ/Q calculation was revised to re-evaluate the offsite χ/Q values, as related to the use of the PAVAN computer model based on finer wind speed categories provided in the latest NRC RG1.23, "Meteorological Monitoring Programs for Nuclear Power Plants;" Revision 1, March 2007. As argued in the question number 3 of the screening criteria, the change is considered a change to an element of methodology. As specified in Reference 3, the NRC staff performed a qualitative review of the inputs and assumptions used in the licensee's PAVAN computer calculations and of the resulting χ/Q values. The staff calculated comparative χ/Q values, and found the results to be similar to the EAB and LPZ χ/Q values calculated by the licensee. On the basis of this review, the NRC staff determined that the resulting offsite EAB and LPZ χ/Q values for the MSLB and SGTR generated by the licensee and presented in Table 3.3.1-6 of Reference 3 SE are acceptable for use in making DBA dose assessments.

"¥" The dose increase is due to change in an element of methodology to correct RADTRAD error in modeling (IR 01320861). The RADTRAD error applies to the LOCA analysis, because this analysis uses modeling method that combines the release into two compartments, representing the sprayed and unsprayed regions within the structure in single run. The LOCA re-analysis uses a separate RADTRAD model for each containment compartments leakage dose contributions (sprayed and unsprayed regions), based on the software issuer suggested modeling method to compensate for the code internal error. The dose results from the two runs were added to determine the total calculated doses associated with the accident.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 400277/EC 399174/DRP 16-012 **Revision Number:** 000 / 000 /000

Title: Revising AST Accident Dose Calculations

Arkansas Nuclear One performed the LOCA dose reanalysis using the exact methodology as above to compensate for the RADTRAD error. As noted in the December 24, 2013 letter (ML13326A502- Arkansas Nuclear One, Units 1 and 2, Safety Evaluation Related to RADTRAD Error) the NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its LOCA radiological dose consequence analyses for ANO-1 and ANO-2 and concludes that they are consistent with the conservative guidance provided in RG 1.183. The NRC staff used the RADTRAD computer code to perform an independent confirmatory dose evaluation to ensure an understanding of the licensee's methods. The NRC staff concludes that summing the results of multiple runs, as described by the software issuer to correct the error, is an acceptable approach for correcting the RADTRAD error because it will account for all the nuclides available for release. The NRC staff evaluated the radiological consequences resulting from the postulated LOCA using the AST and concluded that the radiological consequences at the EAB and LPZ and in the CR are within the dose criteria specified in 10 CFR 50.67. Therefore, this change is acceptable with respect to the radiological consequences of DBAs.

References:

1. Letter from Craig Lambert (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate," dated June 23, 2011 (ML111790030).
2. Letter from N. J. DiFrancesco (U. S. NRC) to M. J. Pacilio (Exelon Generation Company, LLC), "Braidwood Station, Unit Nos. 1 and 2 and Byron Station, Unit Nos. 1 and 2 - Supplemental Information Needed for Acceptance of Licensing Action Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME6587, ME6588, ME6589, and ME6590)," dated August 22, 2011 (ML112150563)
3. Letter from Joel. S. Wiebe (U. S. NRC) to M. J. Pacilio (Exelon Generation Company, LLC), "Braidwood Station, Unit Nos. 1 and 2 and Byron Station, Unit Nos. 1 and 2 - Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC NOS. MF2418, MF2419, MF2420, AND MF2421)," dated February 7, 2014 (ML13281A000).
4. NRC Approval letter dated December 24, 2013-Arkansas Nuclear One, Units 1 And 2- Safety Evaluation Related to Revised Dose Consequences Based on Alternate Source Term (ML13326A502).

Summary of Conclusion:

(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.)

This activity will implement Engineering Changes (EC) 400277 (BYR), EC 399174 (BRW) and UFSAR change DRP 16-012. The above ECs incorporate four (4) revised design analyses determining the radiological consequences using Alternative Source Term (AST) methodology. The potential accidents for which Control Room (CR), Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses have been recalculated for Byron and Braidwood are as followed: (a) BYR04-051/BRW-04-0038-M - Re-analysis of Loss Of Coolant Accident (LOCA) Using Alternative Source Terms, Rev. 5. (b) BYR04-045 & BRW-04-0039-M, "Re-analysis of Control Rod Ejection Accident (CREA) Using Alternative Source Terms", Rev. 4. (c) BYR04-047/BRW-04-0041-M, "Re-Analysis of Fuel Handling Accident (FHA) Using Alternative Source Term", Rev. 4. (d) BYR04-049/BRW-04-0043-M, "Re-Analysis of Locked Rotor Accident (LRA) Using Alternative Source Term", Rev. 4.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC 400277/EC 399174/DRP 16-012 **Revision Number:** 000 / 000 /000

Title: Revising AST Accident Dose Calculations

The revised calculated Total Effective Dose Equivalent (TEDE) is within the limits specified in 10CFR 50.67, and RG 1.183 (as shown in the table above).

The error discovered in the version of RADTRAD that has been used in the AOR described in the applicable sections of the chapter 15 in the UFSAR. The correction to this error resulted in the analyzed dose to be higher (conservative) than those previously calculated. The correction in the RADTRAD code is to a design element. Per NEI 96-07 Revision 1, "In general, licensees can make changes to elements of a methodology without first obtaining a license amendment if the results are essentially the same as, or more conservative than, previous results." As shown in table above, the results are conservative and remained within the limits of 10 CFR 50.67(b)(2) and RG 1.183.

There are no changes by the Activity to the manner in which the plant is operated or controlled. This change is strictly analytical in nature, thus it does not constitute a test or experiment. There are no changes to the Technical Specifications or the Operating License required. Because this activity results in higher doses calculation for the affected accidents due to changes in elements of the methodology the Screening question number 3 is answered. "Yes". The remaining Screening questions are answered "No" the activity screens in and a full 50.59 Evaluation is 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.**	(BYR) 6E-15-043/ BRW-S-2015-74	**Rev.** 0/0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	(BYR) 6G-16-002/ BRW-E-2016-22	**Rev.** 0/0

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

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: Action Tracking Assignment 01493278-02 **Revision Number:** 0

Title: Evaluation of Long-Term Removal from Service of the CVCS Positive Displacement Pump to Address 2013 NRC Inspection Finding at BWD

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 evaluation is being performed to document the acceptability of long term operation with the Chemical and Volume Control System (CVCS) Positive Displacement Pump (PDP) out of service.

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

Byron and Braidwood Stations made the decision early in plant life to not use the PDP for normal CVCS charging functions. The decision was based on the PDP's low capacity which made it unable to support station preferred Reactor Coolant System (RCS) letdown flow rates for coolant chemistry control. The PDP was tagged out of service with no intent of performing maintenance. This constituted a plant configuration change which requires a review in accordance with 10 CFR 50.59. In 1997, Braidwood Station completed a 10 CFR 50.59 Safety Evaluation, BRW-SE-1997-676, to document the long term out of service on the PDP. Evaluation BRW-SE-1997-676 was made applicable to Byron Station in 1998 (Byron Validation #6H-98-0171).

In May 2013, the NRC issued Braidwood Station Integrated Inspection Report 05000456/2013002, 05000457/2013002. The Inspection Report identified a Severity Level IV finding associated with Safety Evaluation BRW-SE-1997-676. The Finding indicated that

> A Licensee 10 CFR 50.59 evaluation performed in 1997 examined a number of indirect consequences, but failed to adequately evaluate the direct consequences that isolating and removing the CVCS PDPs from service would have on the CVCS PDP-supported safety functions as described in the UFSAR...Specifically, the safety evaluation failed to evaluate the direct effect of revising the CLB to permit isolation and removal from service of the CVCS PDP for an extended and undetermined period of time on functions important to safety, including normal and emergency RCS boration, RCS inventory control and reactor coolant pump seal injection.

As a result of the Finding documented in the Inspection Report, Exelon committed to re-perform the Safety Evaluation to ensure the identified concern was addressed (Ref. Braidwood IR 01512303 and Byron IR 01493278).

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

There is no impact on plant operations, design bases or safety analyses described in the UFSAR. Specifically, there is no impact on normal and emergency boration, RCS inventory control, or reactor coolant pump seal injection. This activity is addressing a historic decision to not utilize the CVCS PDP for normal charging duties. Revisions were made to the UFSAR in 1997 to reflect the potential long term removal from service of the PDP (ref. DRP 7-072). Procedures impacted by the unavailability of the PDP have already been revised.

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 Evaluation utilized the statements made in the UFSAR and the NRC Safety Evaluation Report (SER) to identify relevant System, Structure and Component (SSC) Design Functions, Design Bases Functions, Equipment Important to Safety and what constitutes UFSAR descriptive information. The review confirmed that while the PDP did have UFSAR specified functions, it is

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: Action Tracking Assignment 01493278-02 **Revision Number:** 0

Title: Evaluation of Long-Term Removal from Service of the CVCS Positive Displacement Pump to Address 2013 NRC Inspection Finding at BWD

not considered to have any Design or Design Basis Functions and is not Equipment Important to Safety. The CVCS PDP is not credited in any Safety Analysis in the UFSAR for the mitigation of any accident or transient.

- *The PDP is not credited for mitigation of any transient or accident and is not the initiator of any transient or accident. Since the PDP is not the initiator of any accident or transient, the activity does not have the potential to increase the frequency of occurrence of an accident previously evaluated in the UFSAR.*
- *Since the PDP is not credited for the mitigation of any accident or transient, the activity does not have the potential to increase the consequences of an accident previously evaluated in the UFSAR.*
- *Since the PDP is not the initiator of any accident or transient, the activity does not have the potential to create a possibility for an accident of a different type than any previously evaluated in the UFSAR.*
- *The long-term removal from service of the PDP does not affect any DBLFPB since the PDP is not credited for mitigating the consequences of any accident or transient and does not provide a support function for any equipment credited with protection of any DBLFPB.*
- *This long-term removal from service of the PDP modifies the normal system alignment that does not involve a method of evaluation as defined in LS-AA-104. Therefore, the removal of the PDP from service 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 Evaluation performed an in-depth review of the Current Licensing Basis information related to the Design Basis Functions of the CVCS. The Evaluation demonstrated that the CVCS PDP does not have any UFSAR Design Basis Function showing that the PDP was neither (1) required by, or otherwise necessary to comply with, regulations, license conditions, orders or technical specifications, nor (2) credited in safety analyses to meet NRC requirements. Excerpts from the UFSAR and from the NRC SER were used to demonstrate that the Safety Analyses associated with the CVCS credited only the safety related equipment and flow paths in the mitigation of events. It was shown that although a Failure Modes and Effects Analysis Table mentions the possibility of using the PDP for mitigative purposes, this is considered to be descriptive information, not a specified Design Basis Function for the PDP since the actual plant response to the event documented in the Safety Analysis does not credit any response from the PDP.

The Evaluation concluded, based on the Safety Analyses presented in the UFSAR and the statements of acceptance provided in the NRC SER, that the CVCS PDP is not considered to be Equipment Important to Safety since the PDP is neither (1) relied upon to mitigate accidents or transients; nor (2) equipment whose failure could prevent safety-related SSCs from fulfilling their Design Function. The PDP is not relied upon to mitigate any accident or transient. All UFSAR Safety Analyses credit use of the safety related centrifugal charging pumps. While some station non-safety related equipment is credited for mitigating transients (e.g. non-ESF Boric Acid Transfer pumps), the PDPs are not explicitly relied upon. The failure of the PDP would have no effect on any safety related or non-safety related SSC fulfilling its Design Function. No station equipment is dependent on the operational support of the PDP.

Based on this Evaluation, it was determined that the long term removal from service of the PDP does not have the potential to increase in the likelihood of occurrence, or increase in the consequences of, a malfunction of an SSC important to safety previously evaluated in the UFSAR. In addition, the 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 since the documented Safety Analyses did not credit performance of the PDP so the results documented remain unaffected. Thus, the proposed activity may be implemented per plant procedures without obtaining a License Amendment.

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

Forms Attached: (Check all that apply.)

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: Action Tracking Assignment 01493278-02 **Revision Number:** 0

Title: Evaluation of Long-Term Removal from Service of the CVCS Positive Displacement Pump to Address 2013 NRC Inspection Finding at BWD

☐	**Applicability Review**				
☐	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-16-003	**Rev.**	0

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

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: Engineering Change (EC)/EC 406220 & EC 406221 and DRP # 16-079 **Revision Number:** 1,1, 0
Title: Reroute AF Diesel Pump Combustion Air Intake to 364' General Area 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.)

The proposed activity is described in EC 406220 and EC 406221 and is being implemented to reroute the combustion air intake lines of the unit 1 and 2 Auxiliary Feedwater (AF) Diesel Pumps such that the combustion air is drawn from the 364' Elevation of the Auxiliary Building general area near the ceiling. The intake lines will be routed through new floor penetrations at 383' Elevation into the AF Diesel Pump rooms and over into the combustion air intake plenums of the 1(2) AF Diesel Pump engines. In the event that the AF Diesel Pumps are started, combustion air drawn from 364' Elevation of the Auxiliary Building will enter the AF Diesel Pump engines and will be exhausted through lines 1(2)DO85A-16" (Ref. P&ID M-50, Sheet 3 and M-130, Sheet 2) and eventually released out to the atmosphere via the existing vents on the Auxiliary Building roof. In addition, EC 406220 and EC 406221 provide the design details to isolate the existing combustion air intake flow paths from the Turbine Building to the AF Diesel Pump rooms (Line 1(2)DOB1A-14", Ref. P&ID M-50, Sheet 3 and M-130, Sheet 2) and abandon the lines in place.

UFSAR Change DRP # 16-079 is being processed to clarify that with the AF Diesel Pump engine intake removed from the Turbine Building, the location of the Carbon Dioxide Storage Tank is no longer near the Unit-1 AF Diesel Pump engine intake. This is a descriptive change of an equipment location and is not associated with SSC design functions.

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

During the 2016 Braidwood NRC Component Design Basis Inspection (CDBI), the inspection team requested information supporting the acceptability of locating the combustion air intake for the AF Diesel Pump engine in the Turbine Building. During the review of the available documentation it was determined that the documentation did not support operation of the AF Diesel Pump diesel during a High Energy Line Break (HELB) environmental conditions postulated in the Turbine Building. This issue was also applicable to Byron as the design is the same (IR 2636112).

The current combustion air intakes for the AF Diesel Pump engines draw combustion air from the 401' Elevation of the Turbine Building. The combustion air intake lines are currently routed from the Turbine Building location through the L-line wall into the AF Diesel Pump rooms on 383' Elevation of the Auxiliary Building. During a postulated HELB event, steam is expected to fill the current intake space in the 401' Elevation of the Turbine Building such that sufficient combustion air cannot be supplied from 401' Elevation of the Turbine Building. This activity provides combustion air intake flow paths to draw the required combustion air from 364' Elevation of the Auxiliary Building general area and in this manner eliminate the Turbine Building HELB effects on the AF Diesel Pump combustion air intakes. In addition, the existing combustion air intake flow paths from the Turbine Building 401' Elevation to the AF Diesel Pump rooms on Elevation 383' of the Auxiliary Building are isolated by this activity to eliminate potential communication of the Turbine Building and the Auxiliary Building via the abandoned in place combustion air intake lines.

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

Currently the Auxiliary Building Ventilation (VA) system exhausts air from the Auxiliary Building by drawing the air from the general areas through the potentially higher contaminated areas through the VA system filters and exhaust through the station vent stack. This flow is monitored and in the event of high radiation detection in the Auxiliary Building exhaust air duct, the Auxiliary Building charcoal booster fans are started manually. The charcoal filter bypass dampers are closed automatically and the effluents are routed through the charcoal adsorbers (for removal of radioactive particulates and iodine) before being exhausted to the outdoors.

This activity provides combustion air flow paths to draw the required combustion air from 364' Elevation of the Auxiliary Building general area for the AF Diesel Pump engines. This activity does not result in changes to the

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: Engineering Change (EC)/EC 406220 & EC 406221 and DRP # 16-079 **Revision Number:** 1, 1, 0
Title: Reroute AF Diesel Pump Combustion Air Intake to 364' General Area Unit 1&2

performance or operating characteristics of the AF Diesel Pumps and the AF system. Therefore, there is no adverse impact on Auxiliary Feedwater operations. Elimination of the Turbine Building HELB effects on the AF Diesel combustion air intakes enhances the reliability of the AF Diesel Pump. However, the proposed activity will result in air from the 364' Elevation general area of the Auxiliary Building being used for combustion air for the AF Pump Diesel engines and exhausted directly to the outside, bypassing the Auxiliary Building Ventilation (VA) exhaust flow paths. Currently the AF Diesel Pump engine exhausts are not monitored for radioactive releases. Provisions for local air monitor near the AF engine combustion air intakes permit monitoring of the air for the purposes of reporting releases. These provisions are implemented via revision to procedures RP-BY-301-1001 and RP-BY-301-1004 or generation of new procedures (Ref. EC 406220 and EC 406221).

The methodologies and design limits of the design analyses performed to qualify the modified SSCs and demonstrate compliance with the applicable design requirements, standards and the AF and VA system performance requirements are derived from approved design specifications which are consistent with the design and licensing basis of the plant. Therefore there is no adverse impact on the design bases of the AF and VA systems.

The propose change did not result in changes to Safety Analyses. Therefore, there are no adverse impacts in the safety analyses associated with the AF and VA systems.

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 has no adverse impact on the design functions of the AF system since the supplied combustion air quality and quantity via the flow path from the 364' Elevation of the Auxiliary Building general area is acceptable and has no adverse impact on the AF Diesel Pump engine performance. There are no new failure modes introduced and there are no changes to the design basis conditions of the AF system since the structural integrity and performance requirements of the AF system are not adversely impacted by the proposed change. The reliability of the AF Diesel Pump is enhanced by providing a more reliable combustion air supply source from the Auxiliary Building. The new intake is routed in the Auxiliary Building and is not subjected to Turbine Building related adverse conditions such as HELB, internal/external missiles and missiles generated by natural phenomena. The new intake piping has been classified as Quality Group G and Seismic Category I component. This is consistent with the plant's design and licensing basis.

The existing combustion air intake flow path from the Turbine Building 401' Elevation to the AF Diesel Pump room has been isolated via a qualified Auxiliary Building boundary. This abandoned configuration has been evaluated and found to be acceptable. Therefore, the abandoned flow path does not result in adverse impacts in the safety related equipment in the AF Diesel Pump room. The new floor penetrations (1AB-2799C and 2AB-223) constructed at 383' Elevation of the Auxiliary Building has no adverse impact on the Auxiliary Building flood analysis or the Auxiliary Building ventilation barriers since the new penetration is designed to be a flood and ventilation barrier. In addition, the predicted flood level in the Auxiliary Building elevations is significantly lower than the Elevation of the AF intakes (near the ceiling of El 364') (Reference Analysis no. 3C8-1281-001). Therefore, there is no adverse impact on the AF air intakes due to Auxiliary Building flood. Based on the above, the proposed activity will maintain the design functions and performance requirements described in the UFSAR for the AF Diesel Pumps and the AF system. Therefore, the proposed activity does not involve a change to an AF system related SSC that adversely affects an UFSAR described design function.

The performance of the Auxiliary Building Ventilation System (VA) has been evaluated for the condition resulting when an additional 10,000 SCFM combustion air may be drawn from the Auxiliary Building to supply the AF Diesel Pump engines (5,000 SCFM per unit) and the evaluation concluded that the VA system continues to maintain an airflow path within the Auxiliary Building from areas with lesser potential for contamination (general areas) to areas with greater potential for contamination (cubicle areas) where the combustion air is mechanically exhausted. In addition, it has been demonstrated that there are no limitations on the running time of the AF Diesel Pump engine when drawing combustion air from the Auxiliary Building. The new combustion air intake configurations and the associated SSCs have been

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: Engineering Change (EC)/EC 406220 & EC 406221 and DRP # 16-079 **Revision Number:** 1,1, 0
Title: Reroute AF Diesel Pump Combustion Air Intake to 364' General Area Unit 1&2

qualified via design analyses performed using design requirements and methodologies consistent with the design and licensing basis.

Although no physical change are made to the VA exhaust filtration SSCs, the proposed activity will affect the UFSAR described design features of the VA exhaust system of precluding direct exfiltration of contaminated air from the Auxiliary Building during normal operations and following an accident which could result in abnormally high airborne radiation in the Auxiliary Building. UFSAR Table 15.0-7 "Plant Systems and Equipment Credited for Transients and Accident Conditions" lists the equipment assumed available and credited in the specific accident analysis performed. The VA system is not listed in this table. In the specific accident analysis, the VA system is not credited to perform the function of filtering plant effluent for dose reduction to the environs. This activity, by providing a flow path for the general areas in the Auxiliary Building to the environs does not change the assumptions of the dose consequence of any accident evaluated. Although it was determined that the existence of these unfiltered release pathways being created did not impact the radiological dose consequences following a Loss of Coolant Accident (LOCA) and the VA system is not credited with performing any functions during the USAR described accidents, this was determined to be an adverse impact requiring a 10 CFR 50.59 Evaluation. The 10 CFR 50.59 Evaluation concluded that the operation of the AF Diesel Pumps and the VA system were not degraded, the activity has no impact on the potential for or consequences of malfunctions of SSC's important to safety. Also, since the AF and VA systems are not accident initiators, there is no increase on the likelihood of an accident previously evaluated in the UFSAR from occurring. As indicted above, since the proposed activity was evaluated to have no impact on the radiological dose evaluations, it was determined to have no impact on the consequences of an accident previously evaluated in the UFSAR. As failure scenarios for the AF and VA systems are unaffected, the proposed activity does not create the potential for an accident of a different type than previously evaluated in the UFSAR. As the modified SSCs perform a passive function, no new malfunctions of SSCs important to safety with different results than any previously evaluated in the UFSAR were identified. The modified SSC's do not result in operation that can impact the containment, Reactor Coolant System, or fuel cladding, therefore, the proposed activity does not result in a fission product barrier being exceeded or altered. Finally, no methods of evaluation not described in the UFSAR were utilized to evaluate the proposed activity.

The proposed activity does not involve a change to a procedure that performs or control the UFSAR described design functions. The procedures for operation of the AF system and the VA exhaust system are unchanged. The proposed change did not decrease the reliability of the AF Diesel Driven Pump or the VA system design functions or functions whose failure would initiate a transient/accident or functions that are relied upon for mitigation. Therefore, the proposed activity does not involve a change to a procedure that adversely affects how UFSAR-described SSC design functions are performed or controlled.

The proposed activity did not result in changes to the safety analyses. The methodologies and design limits used in the design analyses performed to qualify the modified SSCs and demonstrate compliance with the applicable design requirements, standards and the AF and VA system performance requirements are derived from approved design specifications which are consistent with the design and licensing basis of the plant. These analyses do not involve changes to an element of a UFSAR described methodology or use an alternate evaluation methodology used to establish the design bases of the plant. Based on the above, this activity does not result in a change to an element of a methodology described in the UFSAR nor does it use an alternative evaluation methodology that is used in the design bases or in the safety analyses.

The proposed change to the AF Diesel Pump engine combustion air intake sources and the isolation of the existing combustion air intake flow path from the Turbine Building to the AF Diesel Pump room does not involve a test or experiment. The revised combustion air intake configuration and associated SSCs have been qualified via design analyses consistent with the plant's design and licensing bases. Based, on the above 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.

The proposed activity does not require changes to the Technical Specifications or the Operating License since it does not alter parameters which the Technical Specifications for the AF and VA system are based on.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: Engineering Change (EC)/EC 406220 & EC 406221 and DRP # 16-079 **Revision Number:** 1,1, 0
Title: Reroute AF Diesel Pump Combustion Air Intake to 364' General Area Unit 1&2

Based on this evaluation, the proposed activity requires a 10 CFR 50.59 evaluation; however, the resulting evaluation determined that the proposed activity can be performed without prior NRC permission per the applicable governing procedures.

The Fire Area Analysis and the Safe Shutdown Analysis of the Auxiliary Building general area is being addressed in the Fire Protection Regulatory Review of EC 406220 and EC 406221. An Applicability Review has been included as part of this 50.59 Review.

Changes made to the effluent pathways and the impact of these changes to the ODCM or offsite dose is reviewed under 10CFR20 and is not evaluated under the 50.59 evaluation. See attached Applicability Review.

Attach all 50.59 Review forms completed, as appropriate.

Forms Attached: (Check all that apply.)

☒	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	6E-16-083	**Rev.**	0
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-16-006	**Rev.**	0