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JCST

10 CFR 50.59 Safety Evaluation Form

Tracking No.: 6G-00-0079 Rev. 1 Ultrasonic flow Measurement REV. 1

1. Station/Unit: Byron/1&2 Applicable Modes: All

Other relevant plant conditions: N/A

2. List the documents implementing the proposed change. Include Procedure Number(s), Test(s), Experiment(s), etc., (including revision # as appropriate):

- DCP 9900071(U-1) including DCN 001386I
- DCP 9900073(U-2) including DCN 001387I
- DCP 9900505(SSCR # 00-020) Setpoint Change for Unit 2 S/G nozzle flow high Alarm
- DCP's 9900500,501,502,503(SSCR #s 00-015,016,017,018) Scaling Change for Unit 2 T_{Ave} and Delta T's
- DCP's 9900499 & 504(SSCR #s 00-014 &019) Scaling Change for Unit 1 and Unit 2 Turbine Impulse Pressure
- DCP 9900506(SSCR # 00-021) Scaling changes for Unit 2 T_{Ave} / T_{Ref} from 583 to 582.5 degree F(T_{Ref} Program)
- DCP 9900507(SSCR # 00-022) Scaling changes for Unit 2 T_{ave}/T_{Ref} from 583 to 582.5 degree F(Pressurizer Level Program)
- DCP 9900508(SSCR # 00-023) Scaling changes for Unit 2 T_{ave}/T_{Ref} from 583 to 582.5 degree F(Steam Dump Control Program)
- DCP's 9900495,496,497,498(SSCR # 00-010,011,012,013) Scaling Changes for Unit 1 Delta T's
- Procedure 1BVSR 4.1.4-1 Reactor Coolant System Flow Measurement
- Procedure 2BVSR 4.1.4-1 Reactor Coolant System Flow Measurement
- Procedure 1BOSR 3.1.2-1 Calorimetric Calculation Daily Surveillance
- Procedure 2BOSR 3.1.2-1 Calorimetric Calculation Daily Surveillance
- BVP 800-43 Feedwater Ultrasonic Instrumentation Connection and Disconnection
- BVP 800-44 Feedwater Venturi Calibration U-1 and U-2
- 1/2 BGP 100-3 Power Ascesation REV. 1
- BOP FW-25 Feedwater Flow constant
- BCB-2 Byron Unit 2 Cycle 9 Figures 33, 33a, 34
- Operating Aid- Feedwater Flow Measurement

PEPP-E FORM

B-12

10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev.1

- 1/2 BGP 100-4 Power Descesation
- 1/2BGP 100-4T3 Load Change Instructions Sheet for Power Decreases <15% in 1 Hour
- 1/2BGP 100-4T2 Load Change Instructions Sheet for Power Decreases ≥15% in 1 Hour
- 1/2BGP 100-3T4 Load Change Instructions Sheet for Power Increases ≥15% in 1 Hour
- 1/2BGP 100-3T5 Load Change Instructions Sheet for Power Increases <15% in 1 Hour
- 1/2BOSR NR-1 Power History Hourly Surveillance
- BOP FW-M2 Main Feedwater System Valve Lineup
- BOP HD-7 Returning High Pressure Feedwater Heater 27A/B to Service
- BAR 2-18-E16 C-16 STPT Exceeded
- BAR 2-15-A11,B11,C11,D11 S/G FW Nozzle Flow Alarm
- BISR 3.2.10-200 Surveillance Calibration of S/G Steam Flow/Feed flow Mismatch Protection Set I and II(FW)
- SSP 00-003 Unit 1 AMAG Implementation
- SSP 00-004 Unit 2 AMAG Implementation with T_{Ave} Reduction
- SE 0001 RS 2.2, Requirements Specification for Byron/Braidwood Calorimetric Package

REV. 1

3. Description and effect of proposed activity:

The DCP's 9900071 and 9900073 listed above permanently installed non-intrusive ultrasonic feedwater flow instrumentation upstream of the flow venturis in the Unit 1(2) steam tunnel. The ultrasonic transducers are mounted to brackets that are bolted to the Feedwater piping, one per Feedwater line. A separate 10 CFR 50.59 Safety Evaluation (6G-99-0044) was developed to document the physical installation of the ultrasonic flow instrumentation.

This 10 CFR 50.59 Safety Evaluation documents the acceptability of the use of the ultrasonic flow measurements in the modification of the calorimetric calculations and the revision or development of the procedures listed in Section 2. To allow for this application, the calorimetric calculations will use a correction factor in the determination of the reactor thermal power for both the plant computer and the daily calorimetric calculations. The software for the plant computer has already been modified to accept this correction factor. The calorimetric software for this application has been Verified and Validated.

During reactor operation, discrete reactor power levels are determined on a continuous basis by the plant computer based on inputs received from various monitoring instruments. Of particular importance to the calorimetric calculations is Feedwater flow. The current calorimetric calculation uses the Feedwater flow as obtained from the Feedwater venturis (FE-510, 520, 530 and 540). The

10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev. 1

Feedwater venturis use differential pressure between the upstream tap and that at the throat of the venturi. Industry and plant operating experience has shown that during operation, fouling of the venturis may occur. Also, venturi inaccuracies can be introduced by the difference between the Reynolds numbers in the test loops used to determine correction factors for Byron venturis, and the Reynolds numbers in the feedwater lines during the power operation. Fouling of the venturi results in a reduction of the flow cross-section through the throat of the venturis and an increase in the pressure drop through the throat due to the roughness of the fouling scale. Both of these conditions contribute to an indicated Feedwater flow higher than actual. Indicated Feedwater flows that are higher than actual values result in an overly conservative calorimetric reactor power calculation. To address this overly conservative calorimetric reactor power calculation, it is proposed that correction factors be manually input to the plant computer and daily calorimetric calculations to compensate for the Feedwater venturi fouling and other venturi induced measurement uncertainties. The NRC has issued an SER on March 20,2000 accepting CE topical report for use of cross flow ultrasonic flow measurement to correct feedwater flow measurement due to fouling of venturis and thereby calorimetric calculation and calibration of nuclear instrumentation. Although CE/AMAG report is not available to ComEd, the plant-specific Byron AMAG Instrumentation was installed by the vendor in accordance with the installation requirements of the vendor. The plant specific accuracy results are documented in ABB calculations 059-PENG-CALC-084 Rev. 0 for Byron Unit 1 and 159-PENG-CALC-085 for Byron Unit 2.

Installed under the DCPs 9900071 and 9900073 listed in Section 2 are ultrasonic flow measuring devices manufactured and installed by Advanced Measurement and Analysis Group (AMAG). These ultrasonic flow-measuring devices have a higher accuracy than the differential-pressure venturis and are not affected by fouling and other venturi induced measurement uncertainties associated with the venturi flow element. Periodically, using new procedure BVP 800-44 ultrasonic flow measurements for each Feedwater line will be taken. Ratio between the ultrasonic flow measurements and those obtained from the venturis will be calculated and provided to the Operations department. The Operations department is responsible for manually entering these constants into the plant computer to correct for inaccurate Feedwater flow. The plant computer calorimetric software will calculate reactor power using these constants. This same factor will be used in the daily calorimetric calculations. The correction factor will be used in the plant computer and daily calorimetric calculations until a new factor is developed through subsequent ultrasonic Feedwater flow measurements, or when the Operations Shift Supervisor determines that the use of the correction factor is not appropriate based on plant operating conditions and the guidance provided by Operating Aid and Operating procedure BOP FW-25.

DCP 9900505(SSCR # 00-020) changes Unit 2 S/G nozzle flow high Alarm setpoint to account for increased feedwater flow.

DCP's 9900499 & 504(SSCR #s 00-014 &019) Change Turbine Impulse Pressure scaling for Unit 1 and Unit 2 respectively because of increase in steam pressure due to AMAG implementation.

DCP's 9900500,501,502,503(SSCR #s 00-015,016,017,018) Change scaling and setpoint changes for Unit 2 T_{Ave} and Delta T's. Since the T_{hot} cannot be increased on Unit 2, the T_{ref}/T_{ave} will have to be decreased. This will be done as part of AMAG implementation. NFM has evaluated(Reference: Letter # PSS:00-024) dropping T_{avg} from 583 degree F to 582.45 degree F for Unit 2. This drop in T_{Ave} will allow the delta-T increase from 58 degree F to 59.1 degree F.

DCP 9900506(SSCR # 00-021) changes for scaling for Unit 2 T_{Ave}/T_{Ref} from 583 to 582.5 degree F (T_{Ref} Program).

DCP 9900507(SSCR # 00-022) changes scaling for Unit 2 $T_{ave}/\ T_{Ref}$ from 583 to 582.5 degree F (Pressurizer Level Program).

DCP 9900508(SSCR # 00-023) changes scaling for Unit 2 T_{ave}/T_{Ref} from 583 to 582.5 degree F (Steam Dump Control Program).

DCP's 9900495,496,497,498(SSCR # 00-010,011,012,013) changes scaling for Unit 1 Delta T's.

Following AMAG implementation on Unit 2, the unit will initially indicate about 98.5% reactor power. Since there is only about 0.8% bite in the last main turbine governor valve we will need to re-open the Unit 2 HP FW heater bypass valve (2FW005) to get to 100% reactor power. The process with which to perform this task will be included in the Unit 2 SPP for AMAG implementation. This will require a change to the M-line up for Unit 2 to reflect the OPEN position of this valve.

4. Reason for Proposed Activity:

The proposed activity is undertaken to correct overly conservative reactor thermal power calculations, which result from the Feedwater flow venturi readings that are biased because of fouling and other venturi induced measurement inaccuracies that are associated with the Feedwater venturis. Correction factors will be developed based on Feedwater flow measurements obtained using ultrasonic instruments. The ultrasonic flow measuring devices have a higher degree of accuracy than do the venturis and are not affected by fouling and other venturi induced measurement inaccuracies. Therefore, these measurements can be used to correct the venturi readings (application of correction factors) to obtain more accurate calorimetric reactor thermal power calculations and operate the plant closer to the licensed rating.

5. Review the UFSAR, including Authorized for Use UFSAR changes, Technical Specifications, other relevant SAR documents and Owner-controlled documents and list sections that describe or discuss the affected systems, structures, or components (SSCs) or activities. (Refer to Definition 1.9). List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

Regulatory documents:

10 CFR 50.59 Safety Evaluation Form

Tracking No. 6G-00-0079 Rev. 1

- Use of NUMARC/EPRI Report TR-102348, "Guidelines on Licensing Digital Upgrades", in Determining the Acceptability of Performing Analog-to-Digital Replacements Under 10CFR50.59.
- SER from NRC dated March 20, 2000 on ABB/CE Topical Report CENPD-397-P, Rev. 01, "Improved Flow Measurement Accuracy using Cross Flow Ultrasonic Flow Measurement Technology".
- SER from NRC dated March 1, 2000 for Byron and Braidwood Re-Rack (Byron Amendment 112) based on Holtec Report # HI-982094

UFSAR Sections:

Sect. 3.9	Mechanical Systems and Components
Sect. 4.2	Fuel System Design
Sect. 4.3	Nuclear Design
Sect. 4.4	Thermal and Hydraulic Design
Sect. Attach 4.4.A	
Sect. 6.2	Containment Systems
Sect. 6.3	Emergency Core Cooling System
Sect. 7.2	Reactor Trip System
Sect. 7.7	Control Systems not Required for Safety
Sect. 9.3	Process Auxiliaries
Sect. 10.4	Other Features of Steam and Power Conversion System
Sect. 11.1	Source Terms
Sect. 12.3	Radiation Protection Design Features
Sect. 15.0	Accident Analyses
Sect. 15.1	Increase in Heat Removal by the Secondary System
Sect. 15.2	Decrease in Heat Removal by the Secondary System
Appendix A	Compliance with Regulatory Guides

SER's:

SER-01	Byron SER Section 1 Introduction
SER-04	Byron SER Section 4 Reactor
SER-05	Byron SER Section 5 Reactor Coolant System
SER-06	Byron SER Section 6 Engineered Safety Features
SER-07	Byron SER Section 7 Instrumentation and Controls
SER-10	Byron SER Section 10 Steam and Power Conversion Systems
SER-11	Byron SER Section 11 Radioactive Waste Treatment
SER-15	Byron SER Section 15 Accident Analyses
SER-C	Byron SER Appendix C
SER Supplmnt's	Byron SER Supplements 1 through 8
SER Letters	SER Letters 1998 through 1994

ITS and ITS Bases:

10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev. 1

Sect. 2.1.1	Reactor Core Safety Limits
Sect. 3.1.2	Core Reactivity
Sect. 3.2.1	Heat Flux Channel Factor (F_Q)
Sect. 3.2.2	Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)
Sect. 3.2.3	Axial Flux Difference (AFD)
Sect. 3.3.1	Reactor Trip System (RTS) Instrumentation
Sect. 3.4.1	RCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits
Sect. 3.4.3	RCS Pressure and Temperature (P/T) Limits
Sect. 3.7.16	Spent Fuel Assembly Storage

Other References:

EPRI TR-112118 Nuclear Feedwater Flow Measurement Application Guide

Braidwood 50.59 Safety Evaluation BRW-SE-1999-0802

Byron Safety Evaluation 6G-99-0055 Unit 2 T_{Ref} Change from 581° F to 583° F

AMAG Evaluation Report

NED-0-MSD-8 Sensitivity of B/B Calorimetric Calculations

NED-I-EIC-0233 Daily Power Calorimetric Accuracy Calculation

PSS:00-024 NFS Letter Evaluation of Lower T_{Ave} on Byron Unit 2 for AMAG Implementation

UFSAR Change Log (none)

ZY index was searched using following keywords:

- over-power
- calorimetric
- max power
- 102
- fouling
- reactor power

The keywords defined above resulted in a large number of hits, only the sections, which were pertinent, are listed above. The documents and specific sections of the documents identified above were reviewed and no changes to UFSAR text are required.

10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev. 1

Technical Specifications Bases (RCS Pressure, Temperature and Flow) states that " any fouling that might bias the RCS flow rate measurement greater than 0.1 % can be detected by monitoring and trending various plant performance parameters. If detected, either the fouling shall be quantified and compensated for in the RCS flow rate measurement or venturi shall be cleaned to eliminate the fouling." The AMAG measurement is used to correct the errors/bias associated with venturi measurement and does not necessarily indicate that greater than 0.1 % fouling has occurred. Because AMAG measurement does not use the venturi for measuring flow, successive AMAG measurements may provide another method of detecting fouling. If fouling is detected after the AMAG correction is applied, that fouling shall be quantified and compensated for in the RCS flow rate measurement. This is consistent with the Technical Specifications and therefore the Technical Specifications changes are not required. REV. 1 REV. 1 REV. 1

6. Describe the functions of the affected systems, structures or components.

Feedwater Venturis are differential pressure devices which utilize two sets of instrument taps each of which are provided with dedicated differential pressure transmitters. The Feedwater flow rate is determined as a function of the differential pressure across the venturi. The flow information is a direct input to the plant computer. The plant computer uses this information for the calorimetric calculations.

The Plant Computer accepts input from various plant instruments, including flow information from the Feedwater venturis, and uses this information in the calculation of the reactor thermal power. Key input parameters include feedwater flow, feedwater temperature, blowdown flow, steam pressure, tempering flow for the D5 Steam Generators, etc.

Calorimetric calculations are performed routinely to determine reactor power. Inputs from various plant instruments are used in this calculation. A significant input parameter to the calorimetric calculation is Feedwater flow. The Sensitivity for calorimetric calculations is documented in Calculation NED-0-MSD-8. This calculation also documents the change in reactor power resulting from variation of calorimetric input parameters.

The Nuclear Instrumentation System provides various reactor trip signals for reactor power. The NIS is adjusted using information obtained from the calorimetric calculations of reactor thermal power. The calorimetric power is also used in Peaking factor surveillance, Spent Fuel burn-up, PTLR, Core reactivity surveillance's, Power defect, Xenon history, Preconditioning requirements, ΔT determination, NIS power channel adjustments, S/G duty, and ΔI target determination.

The ultrasonic flow measuring system uses the equipment installed by the DCP's listed in Section 2. The ultrasonic flow measurement uses the sound scattering properties of turbulence rather than using the direct speed of sound travel time via Doppler effect or via phase shifts. Periodically, the permanently installed transducers will be connected to the shared data acquisition/data analyzer and ultra-sonic flow measurements will be obtained. Correction factors will be developed to relate the flow as indicated by the feedwater venturi to that obtained from the ultrasonic measuring system. The correction factors are used in the plant computer calculation of reactor thermal power as well as the daily calorimetric calculation.

7. Describe how the proposed activity will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs. For a test or experiment, discuss the impact on the safe operation of the plant of any new technique or new system configuration.

The use of the ultrasonic Feedwater flow measuring equipment, in-and-of-itself, has no affect on nuclear safety. Periodically, the data acquistion/data analyzer will be connected to the permanently installed transducers, and the Feedwater flow measurement obtained. It is only when this data is used to correct the Feedwater flow as measured by the Feedwater venturis in the plant calorimetric calculations and the determination of reactor thermal power that nuclear safety needs to be addressed.

Currently, the reactor thermal power is determined by using the flow information which is obtained from the Feedwater Venturis. Industry operating history has shown that venturi induced errors/bias of feedwater measurement may occur during each fuel cycle. This results in Feedwater flow measurements higher than actually exist. Indication of higher Feedwater flow than actually exists is due to venturi induced measurement inaccuracies, and conservatively requires that reactor power be reduced resulting in lost power generation.

The use of ultrasonic flow measurement which is not subject to venturi fouling and other venturi induced measurement uncertainties is used to correct the "Actual Reactor Thermal Power". This correction factor will allow the plant operate closer to the 100% rated thermal power. The development and use of the correction factors will be adminisratively controlled to ensure that the reactor will not operate at levels higher than 100% of its rated thermal power as indicated by the calorimetric power. SPP 00-003(U1) and SPP 00-004(U2) will monitor the systems supporting power operation ~~such as Condensate,~~ Condensate Booster, Heater Drains, Feedwater, Circulating Water, Main Steam, Turbine Speed etc., to ensure that the supporting systems for power operations will not

10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev. 1

be exposed to operating conditions beyond their design limits for 100% of unit capacity. The scaling changes and setpoint changes required for AMAG implementation are listed in Section 2 and will be completed as part of AMAG implementation. Based upon plant monitoring during the SPP, additional scaling changes, if required, will be implemented.

Since the T_{hot} cannot be increased on Unit 2 due to S/G degradation, the T_{ref}/T_{ave} will have to be decreased. This will be done as part of AMAG implementation. NFM has evaluated(Reference: Letter # PSS:00-024) dropping T_{Ave} from 583 degree F to 582.45 degree F for Unit 2. The initial change in T_{ref}/T_{ave} will be only -0.5 °F to 582.5° F on Unit 2 and no change on Unit 1. This drop in T_{Ave} will allow the delta-T increase from approximately from 58 degree F to 59.1 degree F. Additionally as listed in Section 2, scaling changes for Impulse pressure on Unit 1 and 2, and S/G nozzle flow high alarm on Unit 2 will be completed as part of AMAG implementation.

Following AMAG implementation on Unit 2, the unit will initially indicate about 98.5% reactor power. Since there is only about 0.8% bite in the last main turbine governor valve we will need to re-open the unit 2 HP FW heater bypass valve (2FW005) to get to 100% reactor power. The process with which to perform this task will be included in the Unit 2 SPP for AMAG implementation. This will require a change to the M-line up for Unit 2 to reflect the OPEN position of this valve.

The Model for Flow Accelerated Corrosion for secondary side will be modified for new flow following AMAG implementation.

The core analysis, The core peaking factor limits, overpower reactor trip, and spent fuel criticality analysis are not affected by AMAG implementation as described in the following paragraphs.

Calculation NED-I-EIC-0233, Daily power Calorimetric Accuracy Calculation Rev. 1, evaluated the impact of using the AMAG ultrasonic flow instrumentation on the 2% RTP error margin. The conclusion is that the use of the AMAG instrumentation would not increase the total error uncertainty above the 2% error margin. In fact, the use of AMAG decreased the amount of error associated with the flow measurement. Since the error does not exceed the 2% margin, the core analysis is satisfied.

The core peaking factor limits are a function of reactor power, with the most conservative limit being applied at 100 % power. An AMAG adjustment to increase the reactor power(which in past has been understated) would indicate that the previous surveillance has applied a slightly conservative Tech Spec limit. Byron Technical Specification Surveillance measurements include a 4% factor to account for uncertainty in the measurement of Nuclear Enthalpy Rise Channel Factor(F^N_{DH}) and 5% factor to account Heat Flux Hot Channel Factor(F_Q) to account for the uncertainty in the measurement of F_Q. Further explanation is provided in Item 19 of this safety eveluation.

OPEX was reviewed for overpower events. No overpower events were caused by erroneous ultrasonic flow measurements.

The assumed uncertainy in the overpower reactor trip (UFSAR Table 15.0.6) includes a 2% assumed uncertainty (1.25% estimated) during the secondary calorimetric method in the calculated reactor power, and an additional assumed 5% axial power distribution (3% estimated) on the axial power distribution effects on total ion chamber current established quarterly using incore/excore calibration procedure. Any change to the incore/excore current due to small change in reactor power will be less than asssumed uncertainty of 2%.

The Byron spent fuel criticality analysis includes 5% uncertainty in the calculated assembly burnups. This is conservative with respect to the 2% reactor power measurement uncertainty, and as assembly burnup is determined as an integral of reactor power, the 2% uncertainty would bound the reactor power being overstated throughout the fuel's operating history. As long as the reactor power and associated integral fuel burnup are established with a method that satisfy the 2% assumed measurement uncertainty, the criticality analysis is satisfied.

When a potentially defouling condition has occurred, based on the guidance provided in Operator Aid and Operating Procedure BOP FW-25, the reactor operator will set the correction factor back to 1.0, nullifying the affect of the ultrasonic flow measurements and request that a new set of ultrasonic flow measurements be taken. Additionally, based upon plant operating parameters, the reactor operator may at any time elect to set the correction factor back to 1.0 (if less than 1.0) to ensure the conservative, reliable operation of the plant.

The feedwater flow correction factor is only applicable to Mode 1 of plant operation. While the factor is present in the plant computer and is used in the daily calorimetric calculation, it has proportionately less impact at lower power levels. Until more experience is gained under partial power conditions, the AMAG flow correction will be set to 1.0 after a significant load drop has occurred.

The ultrasonic flow measurements are taken periodically and there will be times when the data gathering/data analysis equipment will not be available for reverification of the correction factor. Additionally, industry experience does not support a totally uniform behavior of venturi fouling/defouling mechanisms. To provide an additional degree of conservatism, the minimum correction factor permitted, based on the ultrasonic flow measurements, will be limited to 0.98 until sufficient plant specific behaviors are quantified and correction factors less than 0.98 are justified.

If the average correction factor is calculated to be greater than 1.00, all correction factors shall immediately be set to 1.0 and investigated before any further corrections are applied.

Although the plant has been designed to operate at 100% power, it is possible that the previous efforts to optimize plant equipment performance at previous power levels may result in limitations or alarms as power is increased from current full power levels. The most significant of those have been evaluated and are addressed in the implementation SPP's. Other conditions will be addressed as part of the plant monitoring that will occur during the SPP's.

8 Describe all significant permanent or temporary changes to the words and drawings identified in Step 5 resulting from the proposed change described in Step 3. Describe how the facility or procedure will be different than as currently described.

No changes to the wording as presented in the UFSAR is required for the use of the ultrasonic flow measuring system, the use of the correction factors for Feedwater Venturi fouling, the changes in setpoints listed, or the decrease in Tavg in Unit 2 to support the AMAG change.

10 CFR 50.59 Safety Evaluation Form

Tracking No. 6G-00-0079 Rev. 1

NOTE

In some cases, the proposed activity being evaluated may be a candidate for adding words to the UFSAR. Consideration should be given to adding a discussion of key regulatory issues, regulatory documents (Generic Letters, Regulatory Guides, NRC Bulletins, etc.), station commitments and new equipment. (See Regulatory Guide 1.70 for level of detail)

9. Is a permanent change to the UFSAR needed?

☐ YES - UFSAR changes have been initiated via Tracking Control No.: _____

☒ NO -

Proceed to next step

10. Identify each accident or anticipated transient, including LOCA and transient analysis, described in the SAR where any of the following is true:

- The proposed activity alters the initial conditions used in the SAR analysis
- The changed SSC is explicitly or implicitly assumed to function during or after the accident/transient
- Operation or failure of the changed SSC could lead to the accident/transient

The following Accidents/Transients listed below are those that pertain to a feedwater venturi defouling event:

ACCIDENT/TRANSIENT	SAR SECTION
• Operational Transients	Chapter 15.0
a) Step Load Changes	
b) Ramp Load Changes	
c) Load Rejection up to and including design full load rejection	

PEPP-E FORM

- Feedwater System Malfunctions — Chapter 15.1

a) Feedwater System malfunction causing a reduction in Feedwater temperature

b) Feedwater system malfunction causing an increase in Feedwater flow

- Loss of Feedwater — Chapter 15.2

a) Loss of external load

b) Turbine Trip

c) Loss of Condenser Vacuum and other events that result in Turbine Trip

d) Loss of normal Feedwater

Industry experience in the use of ultrasonic flow measuring devices indicates that each of the transients listed above could result in what is termed defouling of the Feedwater venturis. The ultrasonic flow measurements are used to correct for Feedwater venturi fouling and other venturi induced flow measurement uncertainties. In the event of a defouling transient, the use of the correction factor developed from the ultrasonic flow measurements would no longer be appropriate and the correction factor set to unity, removing the effect of the ultrasonic flow measurements until such time as new Feedwater ultrasonic flow measurement data is obtained.

11. May the proposed activity increase the probability of occurrence of any accident or transient, identified in Step 10.

☐ YES ☒ NO

Provide the rationale for the answer for each accident or transient

The proposed activity does not increase the probability of any accidents/transients identified. The use of ultrasonic flow measurements to more accurately determine reactor power by correcting for Feedwater venturi fouling and other venturi induced flow measurement uncertainties will not increase the probability of occurrence of any accident or transient. The ultrasonic flow measurements will be taken periodically; correction factors will be manually input to the plant computer and used in the calorimetric calculations to determine reactor power. There are no control functions or control setpoint features associated with the collection of data, development of the correction factor or use of this factor in the calculation of reactor thermal power. The use of these correction factors does not reduce the reliability of the Feedwater venturis or the plant computer and hence will not result in the probability of occurrence of an accident or transient. Actually, this correction factor will allow the plant operate closer to the 100% rated thermal power. Monitoring of the systems supporting power operation such as Condensate, Condensate Booster, Heater Drains, Feedwater, Circulating Water, Main Steam, Turbine Speed etc., will ensure that the supporting systems for power operations will not be exposed to operating conditions beyond their design limits for 100% of unit capacity. The margin of safety will not be reduced as a result of improving the Feedwater Flow measurement as read by the plant process computer. Since none of these items adversely affects the systems involved in the transients listed in Step 10, the probability of the accident or transient is not increased.

12. May the proposed activity increase the consequences of any accident or transient identified in Step 10.

☐ YES ☒ NO

Provide the rationale for the answer for each accident or transient

Guidance in the form of operating aid and Operating Procedure BOP FW-25 is provided to the Operations personnel for their use in assessing potential transients or events that could result in defouling of the feedwater venturi, and avoid the potential for erroneously operating the reactor above 100% rated thermal power (102% based on calorimetric uncertainties). Maintaining the reactor at 100% of rated thermal power (102% based on calorimetric uncertainties) ensures that the bases for the accident and transient analyses contained in the UFSAR remain valid and are not compromised when the ultrasonic flow measurements are used to correct the reactor thermal power calculations. Maintenance of the accident and transient bases ensures that no increase in the consequences of an accident or transient will occur through the use of the ultrasonic Feedwater flow measurements to correct for Feedwater venturi fouling and other venturi and instrument loop uncertainties.

13. May the proposed activity create the possibility of an accident or transient of a different type than any previously evaluated?

10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev. 1

☐ YES ☒ NO

Provide the rationale for the answer considering the descriptions provided in Steps 6, 7, 8, 11, and 12.

Not currently a defined transient, which is of interest, is a defouling event. Industry experience has shown that certain events may result in defouling of the Feedwater venturi. Among these events are load changes of greater than approximately 10% of rated thermal power, changes in Feedwater temperature decreases greater that 15°F, pH excursions, water hammers, etc. The reactor operators easily recognize these events and in such events resulting in defouling, it is clear that use of the ultrasonic Feedwater flow measurement correction factor must be removed. An event that may not be as readily apparent is a spontaneous, partial defouling that is not accompanied by any recognized defouling mechanism. As such, fouling causes an apparent measured increase in Feedwater flow when in fact no change has occurred. This in turn would require the reactor operator to reduce power. A defouling transient would appear as a reduction in reactor power. This event would appear to the reactor operator as a sudden decrease in reactor power with no decrease in electrical output. Here again, guidance in the form of Operator Aid and Operating Procedure BOP FW-25 is provided to the reactor operator to investigate if the partial defouling event has occurred. As with other defouling events, the reactor operator would cease the use of the ultrasonic Feedwater flow measurement correction factor until such time as the ultrasonic flow measurement data could be recollected and a new correction factor developed.

The defouling is not a new accident or transient. After a defouling event, calorimetric reactor power indication is reduced. Operator action would be required based on the single indication of reactor power (calorimetric) to increase reactor power above its previous value.

14. Describe how the proposed activity will affect equipment failures or malfunctions. Describe any new failure modes and their impact during applicable operating modes and applicable accident or transient conditions.

The use of the ultrasonic flow measurements is merely to correct for inaccurate (overly conservative) Feedwater flow rates obtained from the venturis when fouling and other venturi and instrument loop uncertainties are present. The conservative guidance provided to the reactor operators to discontinue the use of the correction factor in the event that a defouling transient has occurred ensures that the basis for the accident analyses remains valid. The ultrasonic flow measuring equipment is installed external to the Feedwater piping and requires no breach of the pressure boundary. It remains dormant when not in use and is qualified by the vendor for EMI/RFI issues. The AMAG instrumentation will be verified to provide accurate measurement per vendor recommendations prior to its use each time using Station procedures. The physical installation and the use of the correction factor will not affect any equipment failures or malfunctions previously evaluated in the UFSAR.

When the average correction factor is determined to be greater than 1.00 during the ultrasonic test, procedure requires that the correction factors are set to 1.00 and the cause of this condition is investigated. No new failure modes are created. The proposed activity will not affect the operation of the equipment during applicable operating modes and applicable accident /transient conditions (Feedwater system malfunction causing an increase in Feedwater flow in 15.1) remain bounded for these conditions.

15. May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety identified in Step 14?

☐ YES ☒ NO

Provide the rationale for the answer for each malfunction described in Step 14.

The ultrasonic flow measurement system has no direct interface with the plant. The data collection of the ultrasonic flow measurements uses dedicated separate equipment, which has no interface (except 120V AC power temporarily during data gathering) with any plant system, or equipment. The correction factor is manually input and has no control function. Guidance in the form of an Operator Aid and Operating Procedure BOP FW-25 is provided to operations personnel ensure that the factor to correct for fouling of the Feedwater venturi is not used in the event a potentially defouling transient occurs. Cessation of the use of the correction factor, the lack of interface with plants systems, equipment or components and the benign nature of the physical installation ensure that this proposed activity will not increase the probability of a malfunction of equipment important to safety.

16. May the proposed activity increase the consequences of a malfunction of equipment important to safety identified in Step 14?

☐ YES ☒ NO

Provide the rationale for the answer for each malfunction described in Step 14.

As stated in Step 14, there are no equipment malfunctions affected by the use of the ultrasonic flow measurements or the correction factor used to negate the effects of fouling of the Feedwater venturis. There is no direct interface with any equipment important to safety.

17. May the proposed activity create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated?

10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev. 1

☐ YES ☒ NO

Provide the rationale for the answer considering the descriptions provided in Steps 6, 7, 8, and 14.

The ultrasonic flow measurement system has no direct interface with the plant. The data collection of the ultrasonic flow measurements uses dedicated, separate equipment, which has no direct interface with any plant system, component or equipment. The ultrasonic flow measuring equipment is installed external to the Feedwater piping and requires no breach of the pressure boundary, sits dormant when not in use and qualified by the vendor for EMI/RFI issues. The guidance in the form of an Operating Aid and Operating Procedure BOP FW-25 provided to the operations personnel ensure that the use of the defouling correction factor will be stopped in the event that a potentially defouling transient were to occur. Cessation of the use of the defouling correction factor ensures that the 100% reactor thermal power operating limit is not violated. Based on the discussion above, the proposed activity will not create the possibility of a different type of malfunction of equipment important to safety than those previously evaluated.

18. List each Technical Specification where the requirement, associated action items, associated surveillance's, or bases may be affected. To determine the factors affecting the specification, it is necessary to review the SAR, including approved pending UFSAR changes, where the Bases Section of the Technical Specifications does not explicitly state the basis.

Technical Specification	Acceptance Limit(s)/Margin of Safety	SAR Documents & Section
2.1.1, Reactor Core SLs	Figure 2.1.1-1	B2.1.1; UFSAR 4.4, 5.1, 15.0
3.1.2, Core Reactivity	The reactivity balance limit ensures that plant operation is maintained within the assumption of the safety analysis.	B3.1.2; FSAR Chapter 15
3.2.1, Heat Flux Channel Factor(F_Q)	The $F_Q(Z)$ limits must be maintained in Mode 1 to prevent core power distributions from exceeding the limits assumed in the safety limits.	B3.2.1; UFSAR 15.4.8
3.2.2, Nuclear Enthalpy Rise Hot Channel	$F^N_{\Delta H}$ shall be maintained within the limits of the	B3.2.2; UFSAR 15.4.8

PEPP-E FORM

10 CFR 50.59 Safety Evaluation Form

Tracking No. 6G-00-0079 Rev. 1

Factor($F^N_{\Delta H}$)	relationship provided in the COLR. $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the greatest relative heat generation with fixed heat removal capability and thus has highest probability for DNB.	
3.2.3, Axial Flux Difference(AFD)	AFD requirements are applicable in Mode 1 above power level 15% RTP. Above 50%RTP, the combination of thermal power and core peaking factors are the core parameters of primary importance in safety analyses. Between the 15% and 90% RTP, the LCO provides penalty deviation time limits to ensure that the distributions of xenon are consistent with safety analysis assumptions	B3.2.3; UFSAR Section 7.7.1.3.1
3.3.1, Reactor Trip System Instrumentation, Table 3.3.1-1	Safety Margin is incorporated into the Allowable Values for reactor Trip setpoints.	B3.3.1
3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	LCO 3.4.1 requires RCS total flow rate ≥ 371,400 gpm. Accident analyses using the Revised Thermal Design Procedure assume an initial nominal 366,000 gpm RCS flow rate. For accident analyses not using the Revised Thermal Design Procedure an initial nominal RCS flow of 358,800 gpm. These thermal design flow	B3.4.1; UFSAR 4.4, 5.1, 15.0; SER 4.4.1

10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev. 1

	rates are 5.7% - 6.7% lower than actual operational flow rates (best estimate flow).	
3.4.3, RCS Pressure and Temperature(P/T) Limits	Each PTLR provides P/T limit curves for heatup, cooldown, Inservice Leak and Hydrostatic(ISLH) testing and data for maximum rate of change of reactor coolant temperature	B3.4.3; UFSAR
3.7.16, Spent Fuel Assembly Storage	K_{eff} of Spent Fuel Pool will always remain < 1.00 assuming that the pool is flooded with unborated water and less than or equal to 0.95 assuming the presence of 550 PPM soluble boron in the pool for the Joseph Oat Spent Fuel pool Storage Racks, and assuming the pool is flooded with unborated water for Holtec Spent pool Storage Racks.	B3.7.16

19. Does the proposed activity reduce the margin of safety as described in the basis for any technical specification?

☐ YES – Margin of Safety IS reduced.
☒ NO - Margin of Safety is NOT reduced.

Provide the rationale for the answer each Technical Specification.

A Venturi defouling event will result in an actual indicated calorimetric thermal power reading lower than actual.

When the calorimetric correction factor is not installed, subsequent reactor power adjustments following a defouling event will recover thermal power that was not apparent due to fouling (a clean venturi condition).

If the calorimetric software correction factor has been added, based on ultrasonic flow measurements, and a later defouling event occurs at a RTP of 100%, any subsequent manual adjustment based on the RTP reading has the potential for actual thermal power to exceed our operating license limit of 3411 MW_t. Since the Power Range Nuclear Instrumentation (PRNIs) are unaffected by the defouling event, the reactor power increase could bias up the PRNI indicated power and the reactor trip setpoints. This bias would reduce the margin between the trip setpoint and the Allowable Values. Any subsequent PRNI gain adjustment based on a calorimetric surveillance would remove/reduce the unconservatism in the trip setpoints, but would reduce the operation margin to the trip setpoints.

Defouling events are recognized by the fact that they are typically attributed to plant transients. In addition, the plant process computer indication of calorimetric power is very sensitive to any changes in plant conditions that can induce a defouling event. Procedural mechanisms and operator aids are incorporated in the changes introduced under this safety evaluation to assist in the recognition of a potential venturi defouling event. The procedures will direct operator actions to remove the calorimetric software correction factor provided by the ultrasonic flow measurements prior to making any reactor power adjustments. Ultrasonic feedwater flow measurements will be re-performed prior to re-inserting any calorimetric correction factor.

The compensation provided by the ultrasonic feedwater flow measurements compensates for fouling and other venturi measurement inaccuracies. The expected disparity between actual thermal power and indicated thermal power attributed to defouling will not result in exceeding a Safety Limit based on the margin available while operating at 100% programmed T_{Ave}, rated thermal power, and conservative RCS pressures. Since the ultrasonic feedwater flow measurements also compensate for venturi inaccuracies, it is likely that the disparity between actual thermal power and indicated thermal power will be enveloped by this difference and within the 100±2% calorimetric error assumed in accident analyses. Since the procedures and operator aids will provide a mechanism to recognize a defouling event and remove the venturi compensations prior to adjusting reactor power, it is not likely that a Safety Limit will be exceeded. A similar rationale can be provided for the impact of a defouling event on the margin between reactor trip setpoints and their Allowable Values.

The Technical Specification minimum RCS flow rate of 371,400 gpm assumes RCS flow measurement uncertainties and provides a conservative margin for DNB and non-DNB limiting accidents. The measurement of feedwater flow and the uncertainties associated with this parameter including venturi fouling contribute to the measurement uncertainty of RCS flow. The implementation of the data obtained from the ultrasonic feedwater flow transducers could eliminate a portion of this uncertainty, however it does not change the accident analyses thermal design (i.e., the flows assumed in the accident analysis). In addition, best estimate and actual RCS flow rates are 5.7% - 6.7% greater than the thermal design flow rates. Therefore, there will not be any reduction in the margin of safety associated with RCS flow measurement.

The Applicable Safety Analyses section of the bases for Technical Specification 3.4.1 states that any venturi fouling that might bias the RCS flow rate measurement by greater than 0.1% can be detected by monitoring and trending various plant performance parameters. This statement remains true since the ultrasonic feedwater flow measurements are not impacted by fouling and, therefore, provide a mechanism to trend fouling of the venturis. The correction factors applied to the venturi measurement of feedwater flow to compensate for fouling provides this mechanism. The ultrasonic feedwater flow measurements will not eliminate the requirement to detect and evaluate biasing or to perform precision calorimetric, since it is Byron Station policy to inspect and clean the venturis prior to the performance of a precision calorimetric. Therefore, the bases for Technical Specification 3.4.1 are not affected by the proposed changes.

Calorimetric power is not directly used as part of the calculation of peaking factors F_Q and F^N_{DH} (Tech Spec 3.2.1 & 3.2.2). These peaking factors are measures of relative power. Core burnup is used within the calculation, but the AMAG adjustment effect (~2%) would not cause significant changes to peaking factor measurements. During the performance of the flux map, small changes in reactor power are considered in order to normalize traces to each other. Once the relative peaking factors are calculated, the measured value plus uncertainties is compared to a limit which is dependent upon power (relaxed limit at lower power). The AMAG adjustment has no significant influence on this. The conservative direction for a calorimetric bias is for the calculated value to be higher than actual.

Calorimetric power is used to determine the fuel assembly burnup (Tech Spec.3.7.16) Assembly burnup is a credit towards the criticality analysis. The administration limit assumes that there is a 3% burnup penalty taken on each bundle. The basis for this 3% is the combined effects of the ability to measure reactor power and the ability to measure individual bundle relative power. The conservative direction for a calorimetric bias is for the calculated value to be lower than actual. The implementation of the AMAG correction implies that the burnup credit applied to old fuel is slightly biased higher than what an AMAG adjusted power would be.

Calorimetric power is used to determine the vessel fluence (Tech Spec. 3.4.3)

10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev. 1

Fluence calculations and projections are based on relative power distributions and burnup, and burnup is the integral of calorimetric power and time. The conservative direction for a calorimetric bias is for the calculated value to be higher than actual. The implementation of the AMAG correction implies that the fluence is slightly biased higher than what an AMAG adjusted power predicts. This is conservative. The result would result in some margin gains to the fluence, at the percentage equal to the AMAG correction.

Calorimetric power is used to calculate burnup, and burnup is used in the reactivity surveillance (Tech Spec. 3.1.2. However, the purpose of the surveillance is to confirm that burnup related predictions are within tolerance. There are other burnup related values used for tech spec surveillance's such as Estimated Critical predictions (SR 3.1.6.1) and Shutdown Margin (SR 3.1.1.1) among others. The burnup value used for all reactivity and power distributions surveillance's is the same burnup. Therefore, this procedure is partly to validate the burnup-related predictions. There is no conservative direction.

Calorimetric power is used to determine the nominal hot full power nominal delta-T (Tech Spec 3.3.1) and NIS alignment (Tech Spec 3.3.10). However, the nominal value used in the delta-T alignment and NIS alignment is based on the same calorimetric power used to demonstrate compliance with the license limit. A bias in one will result in a bias in the other. The conservative direction is for the calculated value to be greater than the actual.

The Delta-I target is set to the measured delta-I (Tech Spec. 3.2.3) at the measured power. If the power measurement contains a bias, then the administration of the delta-I would be affected by the same bias in such a way that the bias would cancel out.

Therefore, since procedures and operator aids are establish to recognize a defouling event and to initiate actions to remove venturi compensations from ultrasonic feedwater flow measurements, the margin of safety of safety is not reduced for any Technical Specification by power output adjustments made based on ultrasonic feedwater flow measurements.

10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev. 1

20. Answer all of the following for this change:

Yes	No	
☐	☒	An Unreviewed Safety Question was identified in Questions 11, 12, 13, 15, 16, 17, and/or 19.
☐	☒	This evaluation identified the need to change the Technical Specifications.
☐	☒	This evaluation identified the need to create a new Technical Specification.
☐	☒	This evaluation identified the need for other NRC approval.

If any of the above answers is Yes, proceed to the next step. If all answers were "No", NA the next step and proceed to Step 22.

21. Regulatory Assurance (Provide concurrence or instructions on the course of actions to follow)

N/A Date: N/A
Print/ Signature

22. Assign a Safety Evaluation tracking number and write it on page 1 of this form. Signatures below may be obtained prior to assigning a tracking number.

23. For Safety Evaluations that do not involve an Unreviewed Safety Questions, complete Safety Evaluation Summary Form.

24. I have determined that the documentation is adequate to support the above conclusion.

Preparer: Mahendra Shah / Mahendra Shah Date: 5/12/00
Print / Signature

Dept. / Location: SEC-Mod Design / Byron Phone: 2816

Subject Matter Experts Used (Print Names)	Jeff Drowley
Joe Williams	Dave Eder
Don Hilderbrant/Bob Wunder	Tom Roberts
	Dave Neidich

PEPP-E FORM

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10 CFR 50.59 Safety Evaluation Form
Tracking No. 6G-00-0079 Rev. 1

25. I have determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer: R.J. Niederer / RJNiederer Date: 5-12-00
Print / Signature

Dept. / Location: Reactor Engineering / Byron Phone: 3443

26. I agree the Safety Evaluation is ready for use.

Approver: Thomas E Roberts / Tom E Roberts Date: 5-12-00
Print / Signature

Dept. / Location: Design Eng. / Byron Phone: 4001

27. Preparer shall forward a copy of the Summary Form to the 50.59 Summary Report Coordinator and distribute the Safety Evaluation paperwork in accordance with Station Procedures.

Safety Evaluation Summary Form

Tracking No. 6G-00-0079 Rev. 1
Activity : AMAG Implementation

DESCRIPTION:

The proposed activity is undertaken to correct overly conservative reactor thermal power calculations, which result from the Feedwater flow venturi readings that are biased because of fouling and other venturi induced measurement inaccuracies that are associated with the Feedwater venturis. Correction factors will be developed based on Feedwater flow measurements obtained using ultrasonic instruments. The ultrasonic flow measuring devices have a higher degree of accuracy than do the venturis and are not affected by fouling and other venturi induced measurement inaccuracies. Therefore, these measurements can be used to correct the venturi readings (application of correction factors) to obtain more accurate calorimetric reactor thermal power calculations and operate the plant closer to the licensed rating.

The revision 1 of the safety evaluation added additional procedures in list of documents in Item 2 and clarified two sentences in last paragraph of Item 5 of the safety evaluation. This does not impact the conclusions of original safety evaluation.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because:

The proposed activity does not increase the probability of any accidents/transients identified. The use of ultrasonic flow measurements to more accurately determine reactor power by correcting for Feedwater venturi fouling and other venturi induced flow measurement uncertainties will not increase the probability of occurrence of any accident or transient. The ultrasonic flow measurements will be taken periodically; correction factors will be manually input to the plant computer and used in the calorimetric calculations to determine reactor power. There are no control functions or control setpoint features associated with the collection of data, development of the correction factor or use of this factor in the calculation of reactor thermal power. The use of these correction factors does not reduce the reliability of the Feedwater venturis or the plant computer and hence will not result in the probability of occurrence of an accident or transient. Actually, this correction factor will allow the plant operate closer to the 100% rated thermal power.

The ultrasonic flow measurement system has no direct interface with the plant. The data collection of the ultrasonic flow measurements uses dedicated separate equipment, which has no interface (except 120V AC power temporarily during data gathering) with any plant system, or equipment. The correction factor is manually input and has no control function. Guidance in the form of an Operator Aid and Operating Procedure BOP FW-25 is provided to operations personnel ensure that

Safety Evaluation Summary Form

Tracking No. 6G-00-0079 Rev.1
Activity : AMAG Implementation

the factor to correct for fouling of the Feedwater venturi is not used in the event a potentially defouling transient occurs. Cessation of the use of the correction factor, the lack of interface with plants systems, equipment or components and the benign nature of the physical installation ensure that this proposed activity will not increase the probability of a malfunction of equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because:

   The defouling is not a new accident or transient. After a defouling event, calorimetric reactor power indication is reduced. Operator action would be required based on the single indication of reactor power (calorimetric) to increase reactor power above its previous value. As with other defouling events, the reactor operator would cease the use of the ultrasonic Feedwater flow measurement correction factor until such time as the ultrasonic flow measurement data could be recollected and a new correction factor developed.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because:

   As evaluated in safety evaluation, the implementation of AMAG instrumentation to correct feedwater flow measurement will not affect the function, operation or margin of safety for any SSCs required by the Tech Specs. The modification does not involve changes to any parameters upon which the Technical Specifications are based.

Attachment A
Nuclear Safety Significance Independent Review Form
Page 1 of 1

Section A Document Title: 10CFR 50.59 SAFETY EVALUATION FORM

Document #: 66-00-0079 Revision #: 1

Section B

	Y	N	
Does this Safety Evaluation exactly cover issues that the NRC has previously reviewed and approved? (If the answer is 'Y', mark Sections C, D, and E as not applicable and go to Section F.)	☐	☒	If the answer is 'N', go to Section C.

Section C **Check here if this Section is not applicable. (Only if Section B is marked 'Y')** ☐

Will the New, Revised, Changed, or Cancelled action, activity, or evolution described in the reviewed document:

	Y	N
• Result in a modification or change to a procedure affecting Technical Specifications, ECCS, ESF, or PRA risk significant equipment or systems?	☒	☐
• Result in a modification or change to ECCS, ESF, or PRA risk significant equipment or systems?	☐	☒
• Increase the potential for a plant trip or transient?	☐	☒
• Affect or reduce a risk significant function?	☐	☒
• Affect the ability of the operator to assess or control the nuclear safety status of the plant?	☐	☒
• Significantly increase the potential for release of radioactive material to the environment beyond design limits?	☐	☒
• Change the nuclear safety response of the plant to normal evolutions, anticipated operational occurrences, or design basis accidents?	☐	☒
• Affect the qualification or operational characteristics of installed risk-significant, Safety-Related, components?	☐	☒

If any answer to these questions is 'Y,' continue the review, answering the questions in Section D.

If all the answers are 'N', then mark Section D as not applicable and go to Section E.

Section D **Check here if this Section is not applicable.** ☐

If any of the answers to questions in section C are yes, continue reviewing the document and answer if the New, Revised, Changed, or Cancelled action, activity, or evolution described in the reviewed document will:

	Y	N
• Result in a change to ComEd's Reactivity Management policy?	☐	☒
• Change a risk significant function?	☐	☒
• Include a major design change to the plant? (such as a steam generator replacement or power up-rate)	☐	☒
• Include a major change to risk significant plant processes? (e.g. new accident response method or analysis)	☐	☒
• Create a significant additional burden for the Operating staff?	☐	☒
• Require a change to the Technical Specification basis	☐	☒
• Move activities from Off-line to On-line, where the unavailability of safety significant equipment is increased?	☐	☒
• Contain a full 10CFR50.59 Safety Evaluation of a new or revised procedure.	☒	☐

After answering the questions in Section D, go to Section E.

Section E **Check here if this Section is not applicable.** ☐

Does the described activity represent an Unreviewed Safety Question (USQ)? ☐ ☒ Go to Section F

Section F

If Section B is marked 'Y', the reviewed documents do not present a USQ and do not require further NSRB review. .... ☐

If Section E is marked 'N' and either all of Section 'C' or Section 'D' is marked 'N'; the reviewed documents do not present a USQ and do not require further NSRB review. .... ☐

If Section E is marked 'N' and any question from Section 'D' is marked 'Y'; the reviewed documents do not present a USQ but do require further NSRB review. .... ☒

If Section E is marked 'Y', the reviewed documents present a USQ and require further NSRB review. The discovery of a USQ must be documented in the Corrective Action Program and the NSRB Coordinator notified. .... ☐

[signature]	7-3-00
Reviewer's Signature	**Date**