RBG-47776

ATTACHMENT 1

50.59 SUMMARY REPORT

SAFETY EVALUATIONS FOR:

- USAR MAINTENANCE
- TEMPORARY CHANGES / PROCEDURES
- TECHNICAL REQUIREMENTS MANUAL MAINTENANCE & LICENSE AMENDMENT REQUESTS
- OTHER

ATTACHMENT 1

SAFETY EVALUATIONS ASSOCIATED WITH

USAR MAINTENANCE

SAFETY EVALUATION NUMBER:

2015-002
2015-003
2015-004
2016-001
2016-002

I. OVERVIEW / SIGNATURES[1]

Facility: River Bend Station **Evaluation #2015-002 / Rev. #:** 1

Proposed Change / Document: EC 31803 HVK CHILLER CONTROL DIGITAL UPGRADE (PARENT EC) AS REVISED BY ECN 56609

Description of Change:

REASON AND BASIS FOR REVISION 1

During the initial upgrade conducted under EC-31803, the chillers were limited to 1 start every 20 minutes under all circumstances including Accident conditions. The 20 minute limit was consistent with the older analogue chiller controls being replaced and to maintain like for like design, the limitation was maintained. Both old and new chiller controls contain timers that maintain their memory of the last start even if power is removed from the control system. The 20 minute limitation (known as "Start to Start") was imposed to protect the chiller compressors' induction motor from thermal damage due to heat caused by inrush currents during starting.

Although the change made by ECN 56609 will remove the interlock in the chiller logic associated with START to START wait time less than 20 minutes, procedural controls will prevent use of this relaxation, such that only under accident conditions will it be permissible for a chiller to start after having been started in the last 20 minutes. During accident conditions, it is preferred to maintain the safety function first (provide chilled water for control building HVAC thereby maintaining design basis temperatures) and demote motor protection to second tier consideration. By allowing the chiller to re-start if needed in less than 20 minutes from the previous attempt to start is a change from current design limitations. The new Start to Start limitation for Accident conditions is 2 starts within a 20 minute period.

Following modification, there will still be no credible scenario during which any chiller motor will be started in contravention of NEMA Standard MG 1 Section 1-20.43. This standard provides the framework for proper operation of induction motors used in the chiller application at River Bend as follows: Two starts in succession, coasting to rest between starts, with the motor initially at ambient temperature (Cold Start) or one start with the motor initially at a temperature not exceeding its rated load operating temperature (Hot start). NEMA Standard MG 1 is consistent with limitations described by the chiller motor manufactures' induction motor data (reference Spec 216.210) which states the motors are capable of 2 cold starts or 1 hot start. The chiller motor protection, though a second tier consideration, is therefore maintained following ECN-56609 modification.

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

Other components and systems associated with the chiller do not require the 20 minute start to start limitation. Refrigerant, oil, and water systems are unaffected by motor inrush currents. The component that regulates refrigerant circulation through the compressor (guide vane and guide vane actuator) has a Stop to Start time restriction of 150 seconds (reference N996-0107). This time restriction is to allow the guide vane to fully close following a compressor stop before re-starting the compressor. The Stop to Start restriction prevents loading the compressors' motor immediately which would result in prolonged inrush currents as the motor comes up to running speed . The Stop to Start restriction of 150 seconds will not be changed by EC-56609 and remains in effect for all chillers. Note in revision 0 of this 50.59, the Stop to Start had a typographical error of 180 seconds. This typographical error has been corrected in the 50.59. This error has no impact on the actual setting as all field settings are 150 seconds and all documents reflect the 150 seconds other than the original 50.59 documented in EC-31803. Also, a 3 document references had typographical errors and referred to "261" verses "216" in the vendor series 3216.210-085-XXX and 0216.210-085-XXX.

The appropriately annotated sections of this 50.59 evaluation have been changed to accurately reflect changes implemented by EC-56002 which removes the 20 minute start to start limitation imposed by existing design during Accident conditions.

Acronyms Used in this Document

Because of the number of acronyms used, they have been included as an attachment. See Attachment 2 for a complete listing of Acronyms used throughout this evaluation.

Basis for Full 50.59 Evaluation

A review of Regulatory Issues Summary RIS 2002-22, Section 3.2.2 states in part, "Software failure analysis typically involves making qualitative judgments of the dependability of the system or using conservative bounding levels for failure probability, as appropriate...One important question is whether the software has adverse effects on a design function...An adverse effect may be the potential marginal increase in likelihood of failure as a result of introducing the software. For redundant safety systems, this marginal increase in the likelihood creates a similar marginal increase in the likelihood of a common failure in redundant channels...On this basis, most digital upgrades to redundant safety systems should be conservatively treated as 'adverse' and screened in for further evaluation under 10 CFR 50.59 process...."

Based on the Control Building Chiller (HVK) controls being an Engineered Safety Features (ESF) system, and based on the replacement of analog with digital controls in all four chillers (redundant), the change has been screened in for further evaluation under 10 CFR 50.59 as suggested by the Regulatory Issues Summary report.

Questions for the evaluation have also included guidance as outlined in NEI 01-01, Guidelines on Licensing Digital Upgrades, Appendix A.

Purpose and Scope

Control building Chilled Water System (HVK) control panels (HVK-PNL1A, HVK-PNL1B, HVK-PNL1C, and HVK-PNL1D) and various skid mounted instruments interfacing with the chiller control panels are aging and portions of the systems components are obsolete.

This Engineering Change (EC 31803) will replace the aging analog equipment with a new basic digital controls package. The replacement equipment will mimic the function

of the existing equipment, be easily maintainable, and provide better performance monitoring and troubleshooting capability. This is a phased installation which uses a parent (EC-31803) which evaluates design and children which implement the change.

Specific details of each chiller skid will be established and implemented with children ECs that will replace both the control panel and associated skid mounted instruments. The following child ECs will provide specific information to implement the changes for each chiller skid:

- EC 31805, HVK Chiller Control Digital Upgrade HVK-PNL1A
- EC 31806, HVK Chiller Control Digital Upgrade HVK-PNL1B
- EC 31807, HVK Chiller Control Digital Upgrade HVK-PNL1C
- EC 31808, HVK Chiller Control Digital Upgrade HVK-PNL1D

The basic digital controls package consists of an AdaptiView control panel with a UC800 controller, color touch screen monitor, and interfacing skid mounted equipment including hot gas bypass valve actuator, vane actuator, compressor motor current transformers (CTs), compressor motor potential transformers, pressure transducers, and temperature transducers.

Because both reliability and obsolescence are a concern, the EC will also perform hot gas bypass valve replacements on each chiller as the existing parts are obsolete and have never been replaced.

As a result of the enhanced monitoring and data storage capability associated with the new digital controls upgrade, the existing data acquisition system (IHA-PNI1) used to monitor the status of safety controls during chiller compressor startup, operation, and trips will be modified to spare key chiller control logic points as these are no longer required.

The AdaptiView Control Panel and subcomponents are manufactured by Trane. All equipment installed is supplied by Nuclear Logistics Incorporated (NLI) and is dedicated for Class 1E safety related applications. NLI is a qualified Appendix B supplier of safety related equipment and also maintains a listing on Entergy's Qualified Suppliers List (QSL).

Existing HVK System Description

The Control Building Chilled Water System (HVK) consists of four (4) 100-percent capacity, mechanical refrigeration water chillers, four (4) chilled water circulation pumps, and associated piping, valves, and instrumentation. (Ref: G13.18.2.1*18, G13.18.2.1*059).

The existing HVK system interfaces with Control Building HVAC (HVC). The interface is via the chilled water of HVK. HVC uses the chilled water for the cooling coils of the Main Control Room (MCR), Standby Switchgear Room, and Chiller Equipment Rooms air conditioning units to remove heat generated by various pieces of equipment and provide Operations personnel comfort. Chilled Water Pumps, HVK-P1A(B)(C)(D), provide the motive force for moving chilled water through the various air conditioning unit cooling coils as well as the evaporator of the HVK chiller. Control of the chilled water pumps, chilled water loop valves, air conditioning units, and air conditioning unit controls, setpoints, and dampers remains unaffected by the change. (Ref: PID-22-14J, PID-22-14H)

The existing HVK system interfaces with both Normal Service (SWC) and Standby Service Water (SWP). The interface is at the condenser where the HVK refrigerant is cooled by service water flow through the condenser. Flow through the condenser is a closed loop and is regulated by a pressure control valve (HVK-PVY32A(B)(C)(D)) that is opened on increasing refrigerant condenser pressure that allows heated service water to exit the loop and cooler water to enter the loop. This pressure regulating control feature remains unchanged with the digital upgrade. Service water pumps (SWP-P3A(B)(C)(D)) and their control remains unchanged with the digital upgrade. Two temperature probes and one differential pressure sensor will be added to the service water lines to sense inlet and outlet temperature and pressure across the condenser heat exchanger. The information, temperature and pressure, will be used by the AdaptiView in capacity calculations and information display. (Ref: PID-09-10B)

HVK's ability to provide regulated chilled water based on heat load demand and provide protection and control for the chiller itself is the purpose of the HVK chiller controls. A control panel located on each of the four (4) chiller skids monitors the instrumentation and controls chiller operation. The control panel functions to provide regulation of chiller load, protective features to initiate chiller warnings called pre-trips and initiate chiller trips if key parameters are not satisfied. Chiller trips are accomplished by sending a trip signal to the 480V circuit breaker powering the compressor motor. (Ref: 3216.210-085-001C, 0216.210-085-007, 008, and 009)

Regulation is accomplished by throttling the guide vane to allow more or less refrigerant flow through the system. Under low load conditions below that which the guide vane can regulate, the control panel also throttles the hot gas bypass valve open to allow refrigerant to bypass the condenser thus warming the refrigerant and preventing a chiller trip. Finally, regulation can be affected by compressor motor load which overrides evaporator chilled water temperature and will move the guide vanes in the closed direction to bring compressor motor load to 102% of the full load rating. (Ref: 3216.210-085-001C, 0216.210-085-007, 008, and 009)

The control panel also monitors key permissives to allow the chiller to start and run. Control logic is provided to prevent starting a compressor until normal chilled water flow through the evaporator and normal service water through the condenser have been established. An extreme low flow condition stops the chiller compressor. A chiller automatic trip alarm is provided in the Main Control Room. A chiller compressor pre-trip alarm is also provided in the Main Control Room. (Ref: 0216.210-085-007, 008, and 009).

Manual control switches are provided in the Main Control Room for operation of the chiller compressors (HVK-CHL1A, B, C, D), automatic and manual operation of the Control Building chiller condenser service water recirculation pumps (SWP-P3A, B, C, D), and automatic and manual operation of the Control Building chiller chilled water pumps (HVK-P1A, B,C, D). A data acquisition system is provided to monitor the status of safety controls during chiller compressor startup, operation, and trips. Control logic is provided so that the redundant system will start automatically. Automatic trip of chiller compressors or low airflow through an air-conditioning unit trips the associated operating chilled water recirculation pump. Control logic is provided so that low chilled water flow through a chiller automatically starts the redundant system's chilled water recirculation pump. (Ref: LSK-22-12)

Design Basis

The Control Building Chilled Water System (HVK) supplies chilled water to the Control Building Air Conditioning system. HVK removes the heat generated by personnel and equipment in the main control, Standby Switchgear, and Chiller Equipment rooms. Two redundant and independent trains are provided. The redundant trains are physically separated and protected with a barrier. The system conforms to the single failure criterion. HVK is designed to provide chilled water to the cooling coils in the air supply ventilation systems for the Control Building during all modes of plant operation, including DBA conditions. HVK will operate during normal, shutdown, or accident conditions without loss of function. In the unlikely event of loss of Main Control Room air conditioning, plant shutdown can be performed from the remote shutdown panel.

The Control Building Chilled Water system instrumentation and controls assure that the Main Control Room area, the Standby Switchgear room, and the Chiller Equipment room air conditioning units have adequate chilled water during normal, shutdown, or accident conditions. The areas within the Control Building which are served by the affected chillers are:

1. Main Control Room
2. Standby Switchgear Rooms
3. Battery Rooms
4. HVAC Equipment Rooms
5. Cable Vault
6. General Areas

Each train (one 100-percent capacity chiller and one 100-percent chilled water recirculation pump) is capable of meeting the total chilled water demand. During a LOCA, with loss of offsite power, one 100-percent chiller is capable of removing the reduced heat load generated in the Control Building. Chiller capacity is 189 tons. (Ref: G13.18.2.1*18, G13.18.2.1*059)

During normal operation the makeup water for the chilled water compression tank is automatically supplied by the plant makeup water system and water to the chiller condenser is supplied by the normal service water system (Ref: PID-22-14J, PID-09-10B). During emergency conditions (LOCA), high drywell pressure or low reactor water level isolates the nonsafety-related supply of the makeup water system and allows the operator to switch to the Standby Service Water (SSW) system. Also, during LOCA, the chiller condenser is supplied by the SSW system. Status lights providing indication of the motor-driven pumps and the position of the Chilled Water System valves are provided in the Main Control Room.

During loss of off-site power, the pre-selected chiller compressor 1B or 1D starts up automatically in their proper standby bus loading sequences in Division II. In the event the Division II chiller fails to start automatically, the pre-selected chiller compressor 1A or 1C in Division I starts automatically after a time delay. The two not pre-selected chillers, one on each Division, will not start automatically.

If both chillers were to fail or partially fail, partial cooling can be achieved by using the Standby Service Water instead of the chilled water.

The Chilled Water System is designed to Seismic Category I criteria and is connected to the standby ac power supplies (Ref: EE-001AA, EE-001AB). All equipment is located in the Control Building which is a tornado proof, Seismic Category I structure. Three hour minimum fire barriers are located throughout to mitigate the consequences of a fire. All of the penetrations in these barriers are either sealed with a 3-hour rated configuration or have been evaluated to be adequate to withstand the fire hazards associated with the area in which they are installed (Ref: EB-003AC). Piping, coils, pumps, and chillers are built and "N" stamped according to ASME Code Section III subsection ND Class 3(1). Coils support sections are built in accordance with ASME Code Section III subsection NF Class 3(1). (Ref: Specification 216.210)

Accident Mitigation

Per USAR Table 3.11-2, the Control Building Chillers are required to mitigate the following accidents:

1. Steam Line Break Outside Containment
2. Feedwater Line Break Outside Containment
3. Loss of Coolant Accident (LOCA)
4. High Energy Line Break Outside Containment (RWCU/RCIC)
5. Rod Drop
6. Fuel Handling
7. Recirculation Pump Seizure
8. Recirculation Pump Shaft Break
9. Main Condenser Gas Treatment System Failure

Control Panels Replaced

As stated previously, the control panel monitors key parameters and controls regulation. The panels also provide protective warnings, trips, and permissives for the Control Building chiller compressors. The following panels and associated subcomponents (Relays, Pressure Switches, and Temperature Switches) will be replaced with digital control panels:

1. HVK-PNL1A Control Building Chilled Water Control Monitoring Panel and associated subcomponents including relays, pressure switches, temperature switches, gauges, and indications.
2. HVK-PNL1B Control Building Chilled Water Control Monitoring Panel and associated subcomponents including relays, pressure switches, temperature switches, gauges, and indications.
3. HVK-PNL1C Control Building Chilled Water Control Monitoring Panel and associated subcomponents including relays, pressure switches, temperature switches, gauges, and indications.
4. HVK-PNL1D Control Building Chilled Water Control Monitoring Panel and associated subcomponents including relays, pressure switches, temperature switches, gauges, and indications.

Skid Mounted Instruments Replaced

The following skid mounted instruments listed below will have to be replaced due to incompatibility with the new digital panels. All will retain their function unless noted otherwise.

1. HVK-RTD63A (B)(C)(D) - High Bearing Temperature Detector monitors compressor bearing oil temperature.

2. HVK-TS67A (B)(C)(D) - Compressor Oil Heater Temperature Switch turns on and off to allow the heaters to energize and de-energize. Currently this temperature switch initiates a contactor which carries the full load of the heater. In the new design, the temperature switch will carry the full load of the heater.
3. HVK-TS69A (B)(C)(D) - Evaporator Entering Water Temperature Switch for Hot Gas Bypass initiates a valve open signal on hot gas bypass if existing chilled water temperature is too cold which bypasses refrigerant around the condenser and directs the refrigerant directly to the evaporator. Opening the bypass valve allows the chiller to stay online during low load conditions, below which the vane actuator of the compressor can not regulate temperature.
4. HVK-TC74A (B)(C)(D) - Chilled Water Temperature Control Thermistor monitors chilled water temperature on the outlet of the evaporator and sends signals to the control panel which varies guide vane opening / closing.
5. HVK-TS71A (B)(C)(D) - Chilled Water Low Temperature Switch monitors chilled water temperature exiting the evaporator and provides a pre trip annunciation for a low temperature and compressor trip for a low low temperature.
6. HVK-TS44A (B)(C)(D) - Evaporator Refrigerant Temperature Switch monitors evaporator refrigerant temperature and provides a pre trip annunciation for a low temperature and compressor trip for a low low temperature.
7. HVK-TI36A (B)(C)(D) - Condenser Refrigerant Outlet Temperature Indicator is an indication only gauge located between the condenser and economizer. It will be replaced by an instrument that provides the condenser refrigerant outlet temperature to the new digital control panel.
8. HVK-RTD59A (B)(C)(D) - Compressor Refrigerant Discharge Temperature Detector monitors refrigerant discharge temperature and provides signal to control panel which trips the compressor on high high refrigerant discharge temperature.
9. HVK-RTD61A (B)(C)(D) - Compressor Refrigerant Discharge Temperature Detector monitors refrigerant discharge temperature and provides signal to the control panel which actuates a pre-trip annunciator on high refrigerant discharge temperature.
10. HVK-TS57A (B)(C)(D) - Compressor Motor Winding Temperature Switch monitors compressor motor winding temperature and trips the compressor motor on a high winding temperature.
11. HVK-PS55A (B)(C)(D) - Compressor Oil Pressure Switch monitors compressor oil supply and reservoir differential pressure and trips the compressor on decreasing low low pressure.
12. HVK-PDS56A (B)(C)(D) - Compressor Oil Differential Pressure Switch monitors compressor oil supply and reservoir differential pressure and actuates a pre-trip alarm on low oil pressure.

13. HVK-PS75A (B)(C)(D) - Condenser Refrigerant Pressure Switch monitors condenser refrigerant pressure and trips the compressor on high high refrigerant pressure. Currently this is installed in the control panel but will be relocated to a skid mounted position.
14. HVK-PS76A (B)(C)(D) - Condenser Refrigerant Pressure Switch monitors condenser refrigerant pressure and actuates a pre-trip alarm on high refrigerant pressure. Currently this is installed in the control panel but will be relocated to a skid mounted position.
15. HVK-PI52A (B)(C)(D) Evaporator Refrigerant Pressure Indicator is indication only and provides local indication pressure of the evaporator. Currently located on the evaporator, this will be replaced by an instrument that provides a pressure signal to the new digital control panel.

Skid Mounted Instruments Added

The following skid mounted instruments listed below will have to be added based on the new digital panel controller requirements. These are:

1. Evaporator (HVK-CHL1A (B)(C)(D)-EV1) Chilled Water Flow Differential Pressure. Currently, no such process monitoring exists on the inlet and outlet Class 3 ASME piping to the evaporator. As such, holes will have to be drilled, isolation valves and differential pressure instruments installed, and conduit routed to the new control panel.
2. Condenser (HVK-CHL1A (B)(C)(D)-CND1) Service Water Flow Differential Pressure. Currently, no such process monitoring exists on the inlet and outlet Class 3 ASME piping to the condenser. As such, holes will have to be drilled, isolation valves and differential pressure instruments installed, and conduit routed to the new control panel.
3. Condenser (HVK-CHL1A (B)(C)(D)-CND1) Service Water HX Entering and Exiting Temperature – Two (2) thermo-well installations on entering and leaving chilled water piping. This piping is Class 3 ASME piping. Stress analysis were performed for the modified piping and found to be acceptable. The new instruments also require new conduit routed to the new control panel but can share conduit with other new Condenser Service Water Flow Differential Pressure instrumentation.
4. Compressor Motor (HVK-CHL1A(B)(C)(D)) Instrument Potential Transformers (480 to 30V) – Three PTs will be added to monitor incoming 480V line voltage to the compressor motors. This voltage will provide input to the new control panel for voltage monitoring and capacity calculations. These new PTs will require a new instrument panel (CT/ PT panel) and conduit located new the skid of each chiller.

5. Compressor Motor (HVK-CHL1A(B)(C)(D)) Instrument Current Transformers (275 to 0.1 Amp) – Three CTs will be added to monitor incoming 480V line currents to the compressor motors. This current will provide input to the new control panel for current monitoring, limit functions, and capacity calculations. These new CTs will require a new instrument panel (CT/ PT panel) and conduit located new the skid of each chiller. The new CTs monitor all 3 phases of current whereas the existing CT monitored a single phase of current and was located in the standby switchgear cubicle for the associated circuit breaker.

Skid Mounted Actuators Replaced

Because of the digital upgrade, existing Hot Gas Bypass valve Actuator (HVK-CHL1A(B)(C)(D)MOV1) and Compressor Guide Vane Actuators (No component ID exists in Asset Suite) are incompatible and will be replaced with new actuators. The new actuators will interface with the new digital control panel.

Skid Mounted Valve Replacement

Based on obsolescence the Hot Gas Bypass Valve (HVK-CHL1A(B)(C)(D)MOV1) will be replaced with an equivalent non-obsolete valve.

Equipment Removed

The following equipment will be removed:

1. IHA-PNL1 Control Building Data Acquisition System Panel chiller data points. This panel provides first hit indications for the Control Building chillers in addition to other parameters that will remain unaffected. The panel currently receives data points by all four (4) chillers HVK-CHL1A (B)(C)(D).

   The following chiller control panel indications will be eliminated from interfacing with the plant IHA-PNL1 Data Acquisition Panel and their function will be replaced by the new digital control panel. The indications provided will no longer be available on the Data Acquisition Panel and will be replaced locally at the chiller control panels by improved digital control panel indications, data storage, and trend data capability.

   a. Compressor Impeller Displacement
   b. Compressor Motor Winding Temperature
   c. Compressor Bearing Oil Temperature
   d. Compressor Refrigerant Discharge Temperature
   e. Cooler Refrigerant Temperature
   f. Condenser Refrigerant Pressure
   g. Compressor Motor Starter Overload

h. Cooler Water Flow
i. Condenser Water Flow
j. Compressor Oil Pressure
k. Compressor Oil Pump Operation
l. Cooler Chilled Water Temperature

2. HVK-PNL10A (B)(C)(D) – Chilled Water Compressor Oil Heater Control Relay Panel. This panel currently provides a high current rating contactor to energize and de-energize the compressor oil reservoir heater element. Given the new digital panel has a high current carrying capacity temperature switch the panel is not longer needed and will be removed from the plant.

Summary of Evaluation:
The evaluation of the Control Building Chiller controls digital upgrade has determined the prior NRC approval is not required. This is based on the fact that the new controls provide the same functionality as the existing controls and do not introduce failure modes beyond those already evaluated and approved. The existing safety evaluation of the USAR assumes a complete loss of control building chiller function is not possible due to the four (4) redundant chillers, two (2) independent trains, each with 100% capability of satisfying the design basis function. The new design retains the four (4) chillers, two (2) trains, and 100% capability thus retaining the same improbable loss of Control Building chiller function. Divisional interaction remains unchanged.

In addition, NEI 01-01 guidelines were utilized to evaluate the digital aspects of the design. The NEI 01-01 evaluation concluded the digital features do not introduce unanswered safety questions. In addition, specific attention was applied to the redundant digital aspect for common mode failures. Through evaluation, it is concluded that common mode failures are as improbable on the new design as they were on the existing design as both used common components across both divisions. While the digital equipment is recognized as being more susceptible, it was concluded that based on unsynchronized isolated controls, thoroughly vetted through Validation and Verification, Factory Acceptance Testing, and Operating Experience, in addition to a phased implementation, the reasonable assurance necessary to mitigate common mode failure concerns was provided.

Is the validity of this Evaluation dependent on any other change? ☐ **Yes** ☒ **No**

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ **Yes** ☒ **No**

Preparer: David Anders / See AS / DP Engineering / Engineering / See AS
Name (print) / Signature / Company / Department / Date

Reviewer: R. G. Finkenaur, III. / See AS / DP Engineering / Engineering / See AS
Name (print) / Signature / Company / Department / Date

OSRC: SEE ASSET SUITE FOR SIGNATURE/DATE
Chairman's Name (print) / Signature /

Date 2012-012 (Rev 0), 2015-002 (Rev 1)
OSRC Meeting #

II. 50.59 EVALUATION

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR?

☐ Yes
☒ No

The replacement of the Control Building Chiller components does not introduce the possibility of a change in the frequency of an accident. The USAR accident analysis discusses frequency in section 15.0.3.1. Incidents of moderate frequency occur once in 1 to 20 years and are considered anticipated operational transients. Infrequent incidents occur once in 20 to 100 years and are considered abnormal (unexpected) operational transient. Finally, there are limiting faults which are not expected to occur but are planned for and are considered design basis accidents (DBA).

BASIS:

The Control Building Chillers are not an initiator of any accident as defined in Chapter 15 of the USAR therefore no new failure modes are introduced. In both existing and new chiller control upgrades, interface with existing Main Control Room controls and indication, Standby Switchgear and MCC controls and indication, and Remote Shutdown Panel controls and indication are unchanged. Current Transformers (CTs) located in the Standby Switchgear area are no longer needed and will be spared in place. Data Acquisition, IHA-PNL1, interfaces will be eliminated as the new control panels functionally replace and improve this capability.

Design Discussion

Exiting Design Plant Interface

In the existing design, chiller controls receive start signals from the Main Control Room in the form of a manual start or auto start signal. Manual starts are operator driven actions while auto starts are based on low chilled water flow on the alternate division. The start signal is sent to the chiller control panel which performs safety checks and starts lube oil (signal sent to Standby Switchgear Room MCC) for pre-lubrication. Once the chiller controls are satisfied that it is safe to start(oil pump on, evaporator and condenser flow established, etc...), a signal is sent from the control panel to the Standby Switchgear to close the compressors circuit breaker starting the chiller. The control room also provides flow indicating contacts for both evaporator and condenser water flow to the chiller control panel for proof of flow permissives integrated into the chiller control logic.

Current Transformers (CTs) located in the switchgear provide information to the control panel which provides an override on guide vane position (normally controlled by chiller leaving water temperature) which will limit guide vane position (chiller load) based on current.

Alarms for pre-trip conditions (alarms in which the chiller remains running), are sent directly from the chiller control panel to the Main Control Room.

When a chiller control panel safety parameter is violated, the chiller control panel sends a trip signal to the Standby Switchgear circuit breaker to open. This in turn generates a trip of the chiller. The trip alarm in the Main Control Room is a function of the Standby Switchgear circuit breaker status and not the control panel.

The Data Acquisition Panel captures contact positions of the chiller controls at the time of the trip and maintains these for further evaluation to determine the cause of the chiller trip. These are on/off values and provide no process parameter data.

The Remote Shutdown panel interface allows Operations to control the Division I chillers during a fire in the main control room. Division II chillers are not included in the remote shutdown control circuitry.

New Design Plant Interface

The new design utilizes the existing design interface with exception to Current Transformers and Data Acquisition Panel. Current transformers in the existing design were incompatible therefore the existing CTs were spared in place and new CTs installed locally at the chiller compressor skid. Data Acquisition Panel indications were no longer needed as the new design captures and stores status information, trips, pre trips, and other trend data which both includes existing data acquisition parameters and exceeds these parameters with the addition of process data trending and history logs.

Plant Interface Conclusion

The functions of the interface controls or consequence of their failure remain unaffected. Excluding aforementioned chiller skid interfaces, no new plant interfaces were created with the upgrade therefore fire risk for plant interface circuits / logic remains unaffected if not reduced due to the elimination of Switchgear CT wire and First Hit interface wire. Only local control panels and skid mounted instrumentation are affected in any significant way.

FMEA Evaluation

Failure modes as outlined in USAR Section 9.2.10.3 Safety Evaluation and attached under a separate cover were reviewed. The failure mode considered in the original failure modes analysis was a complete loss of the Control Building chiller control panel; therefore, the existing failure modes and effects analysis bounds any new failure modes introduced by the upgraded chiller control panel.

NEI 01-01

a. Does the new equipment installed with the upgrade exhibit performance characteristics, or have design features, that give an increased frequency of a system malfunction resulting in an accident?

No. Operating experience of the Trane AdaptiView controls and associated sensors indicate reliability is equal to or better than previous controls. The valve replacements are like for like and therefore no increase in malfunction exists. The UC800 controller provides enhanced control functions that increase the likelihood the chiller will remain online and ensure the chiller compressor will operate at optimum efficiency. The AdaptiView monitors process parameters and self calibrates instruments. AdaptiView provides for enhanced monitoring of both chiller parameters and instruments monitoring the processes.

The chiller controller uses soft loading except during manual operation. Soft loading maintains the current of the compressor motor at a specified ramp rate from 40% (established just after the motor exits inrush ~ 6 seconds) to 100% full load amps. This feature ensures that overshoot is prevented during pulldown (maximum estimated temperature to desired temperature difference) and thus eliminates potential chiller trips which on earlier chiller designs were caused by a rapid unloading to compensate for the overshoot. Soft loading therefore allows previously large vane position adjustments due to load or setpoint changes to be made gradually based on load conditions and prevents overshoot which protects the compressor from cycling unnecessarily.

The controller monitors chiller refrigerant temperatures, refrigerant pressures and electrical phase imbalances (<30% limit, $\geq$ 30% trip) and adjusts chiller operation when conditions approach alarm limits. An example of such a condition is when there is a partial failure of a cooling tower, limiting total capacity. These adjustments by the controller under adverse conditions maximize the ability to keep the chiller running. The controls would automatically adjust for these conditions and maintain the chiller online reducing chiller trips whereas the old chiller would trip offline.

In addition, the new control system as modified by ECN-56609 allows 2 starts within 20 minutes under Accident conditions. This function is automatically executed and requires no operator action. Under normal conditions, procedure administrative controls limit the number of starts within 20 minutes to 1. By allowing 2 starts within 20 minutes under Accident conditions, the probability of a complete loss of Control Building chillers is reduced.

The new controls are capable of measuring entering and leaving water temperature, oil temperature, chiller capacity in tons, power consumption, power factor (uncorrected), compressor phase amps, and compressor phase voltage. In addition, data logs include ASHRAE 3 reports (parameters deemed critical by the American Society of Heating, Refrigerating and Air-Conditioning Engineers), custom reports, graphical custom historical data logs, purge reports, and a 50 alarm log which are retrievable from the AdaptiView display screen. These features allow users to monitor and diagnose chiller operation trends.

In summary, the new features tend to increase chiller system reliability thereby reducing the frequency of system malfunction.

b. Does the system exhibit performance characteristics that increase the need for operator intervention or increase operator burden to support operation of the system in normal or off-normal conditions?

No. The new controls act as those of the previous control system with no additional interaction required. The new control panel will enhance the capability of the chiller to remain online (as mentioned above) by adding additional monitoring points and allowing for control in off-normal conditions to compensate and prevent chiller trips. By increasing the availability (operating at or above 158.98 tons capacity without tripping), reliability is increased. By increasing the system reliability, operator burden is decreased. Note that 158.98 tons required cooling capacity for LOCA and 143.39 tons required cooling capacity for normal operations per calculation G13.18.2.1*059.

c. Is the system compatible with the installed environment (e.g., temperature, humidity, seismic, electromagnetic fields, airborne particulates) such that the system performance will not be degraded compared to the system being replaced?

Yes. The new equipment is tested for both noise emissions and susceptibility in accordance with purchase specification SPEC-11-00001-R which invokes EPRI TR-102323, Guidelines for Electromagnetic Interference Testing in Power Plants. Qualification Report 6216.210-085-010 documents the qualification testing of the equipment. The new equipment is housed in an NEMA-12 enclosure which is constructed (without knockouts) for indoor use to provide a degree of protection to

personnel against access to hazardous parts; to provide a degree of protection of the equipment inside the enclosure against ingress of solid foreign objects (falling dirt and circulating dust, lint, fibers, and filings); and to provide a degree of protection with respect to harmful effects on the equipment due to the ingress of water (dripping and light splashing).

The new equipment is seismically tested in accordance with IEEE 344-1975, NRC RG 1.100, Rev.3 and SPEC-11-00001-R and will not affect surrounding equipment during seismic events. Heat introduced by the equipment is reduced from that of the original equipment as the new equipment operates using low powered digital components whereas the old equipment replaced uses traditional high powered analog equipment.

The chiller controls are located in the Control Building Chiller Equipment Rooms used for no other purpose than for the Control Building Chilled Water System mechanical and controls equipment. The Control Building Chiller Equipment Rooms are maintained as Mild Environmental Zones during Normal and Accident conditions thereby establishing a compatible environment for the new control systems.

d. Can the system have an adverse impact on the installed environment (e.g., temperature, humidity, seismic, EMI/RFI emissions, airborne particulates) such that performance of an existing system used for accident detection will be more than minimally degraded compared to existing requirements?

No. The new equipment is tested for both noise emissions and susceptibility in accordance with purchase specification SPEC-11-00001-R which invokes EPRI TR-102323, Guidelines for Electromagnetic Interference Testing in Power Plants.

Testing indicated the guide vane actuator exceeds levels of EPRI TR-102323 at 1 meter but passed (emissions under levels specified by EPRI) at 2 meters from the actuator. A review of the area where the actuator is located (Chiller Equipment Room) indicates there are no equipment (digitally sensitive) within the two (2) meters that are affected by the emissions. Furthermore the new controls are unaffected by the emissions as demonstrated during factory acceptance testing in which the control panel was not enclosed and wiring for sensors was not shielded. In the final configuration, both control panel, sensors, and conductors will be shielded providing further assurance of emissions immunity. (Ref: 6216.210-085-010, 6216.210-085-015)

The new equipment is not an emitter of particulates and adds no humidity. The new equipment is seismically tested in accordance with SPEC-11-00001-R and will not affect surrounding equipment during seismic events. Qualification Report 6216.210-085-010 documents the qualification testing of the equipment.

Heat introduced by the equipment is reduced from that of the original

equipment as the new equipment operates using low powered digital components whereas the old equipment replaced uses traditional high powered analog equipment. All newly installed components are powered from the new panel power source. This includes the control panel and internals, pressure sensors, temperature sensors, hot gas bypass valve actuator, and vane actuator. Heat loads are tabulated in calculation E-225. In the existing design, the total watts generated is based on the panel rating (conservative) and not on actual loading (less conservative) therefore the change is bounded by existing calculated heat load generated and is bounded by the existing calculation.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR? ☐ Yes ☒ No

The Control Building Chiller upgrade does not physically alter any equipment, system performance, or operator actions that could promote the loss of Control Building air conditioning such that the current UFSAR analyses remain bounding for the modes/conditions for which loss of Control Building air conditioning is applicable.

BASIS:

The likelihood of a malfunction occurring of a Control Building chiller was evaluated in the USAR. USAR 3.11.4 evaluates if a loss of Control Building ventilation and subsequently the loss of Control Building chilled water is possible. This same section concludes a total loss of Control Building chilled water is not considered a credible event. However, section 7.4 of the USAR states "In the unlikely event of loss of Main Control Room air conditioning, plant shutdown can be performed from the remote shutdown panel". In addition, River Bend Station has contingency for such an event as evidenced by procedure AOP-0060, Loss Of Control Building Ventilation, which addresses a complete loss of Control Building Chilled Water. In addition, analysis have been conducted as to the severity of such a loss of chilled water and ventilation over time and is documented in calculation G13.18.12.3*161, which calculates area temperatures after 24 loss of all Control Building Chilled water and HVAC. Finally, failure of the Chilled Water System has a diverse means of supplying partial cooling as outlined in SOP-0066, Control Building HVAC Chilled Water System section which allows Service Water to cool the Control Building chilled water loops.

Given a complete loss of Control Building chillers is considered unlikely, an evaluation (NLI FMEA N996-0108 EC31808) was conducted to determine if the new control system changes this conclusion. The new controls upgrade functionally is a like for like replacement of the control panel and instruments with similar control panel and instruments with the exception of the Start to Start restriction changed for the new control system by EC-56609 which allows 2 starts within 20 minutes under Accident conditions. This function is automatically executed and requires no operator action. Under normal conditions, procedure administrative controls limit the number of starts within 20 minutes to 1. By allowing 2 starts within 20 minutes under Accident conditions, the probability of a complete loss of Control Building chillers is reduced. The main distinguishing factor of the new control system is the software that has been introduced and new interface to the local controls which are different (touch screen).

The introduction of software does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR. The likelihood of a software malfunction occurring may be treated the same as that of a hardware malfunction which is currently analyzed for the Control Building chillers. The Chilled Water System supplies chilled water during normal, shutdown, and DBA

conditions to the cooling coils in the Main Control Room air conditioning units, in the Standby Switchgear room's air conditioning units, and in the Chiller Equipment room's air conditioning units without loss of function. Two redundant and independent trains are provided (Ref: USAR 9.2.10.2, 9.2.10.3).

The basis for reliable software is established during verification and validation testing (ref, N996-0109) conducted by the Appendix B qualified supplier, Nuclear Logistics Incorporated (NLI). The V&V program has been performed in accordance with EPRI TR-106439 and IEEE 7-4.3.2-2003 Annexes C and D. Software Quality Assurance Plan has been performed in accordance with the applicable requirements of IEEE 1012-2004 and the RBS specification. Human Factors Engineering (HFE) review of the human-system interface in accordance with the applicable recommendations of IEEE 1023-1988 and NUREG 0700. Requirements Traceability Matrix meets the intent of IEEE 830-1998. Software configuration management plan has been done in accordance with the applicable requirements of IEEE 828-2005. Failure Modes and Effects Analysis (ref. N996-0108, NLI FMEA) has been performed and is equivalent to the recommendations of IEEE 352-1987.

NEI 01-01

a. Does the modified system meet the required plant environmental and seismic envelopes?

Yes. The new equipment was purchased with technical and quality requirements as specified under purchase specification SPEC-11-00001-R which delineates those requirements. Specifically, IEEE 323-1974 and IEEE 344-1975 have been invoked as the qualifying standards for seismic and environmental conditions. Also, specific environmental conditions were delineated per the Environmental Design Criteria database, 215.150. The seismic response spectrum was specified in the purchase specification SPEC-11-00001-R.

The Environmental Qualification program is unaffected by the change. The chillers are located in a mild EQ environment for temperature and radiation. Chiller controls operate in a normal ambient temperature of 40°F to 104°F and relative humidity of 20-90%. The total 40 year integrated dose rate for this area is 800 Rad/(40 years) with a dose rate of 2x10E-3 Rad/hour for normal operation. (Ref: 215.150, SDC-402/410)

The only large lines in the chiller rooms are Chilled Water and Service Water for use by the chillers. The pipe chase in Control Building is separate and does not affect the chillers. The only lines located in the chiller rooms are those that support chiller operation (HVK & SWP). Any rupture in a chiller room would likely cause the division of chiller with the rupture to become inoperable. Therefore, there is no impact on the chillers with respect to Moderate or High Energy Line Breaks (MELB, HELB).

b. Could the environment in which the upgraded equipment operates cause an increase in the likelihood of failure (e.g., electromagnetic susceptibility in a higher frequency range)? Could the new system create an environment (e.g., temperature, humidity, seismic, EMI/RFI emissions, airborne particulates) which adversely affects other equipment and increases the probability of occurrence of a malfunction?

No. The new equipment is tested for both noise emissions and susceptibility in accordance with purchase specification SPEC-11-00001-R which invokes EPRI TR-102323, Guidelines for Electromagnetic Interference Testing in Power Plants. The new equipment is not an emitter of particulates and adds no humidity. The new equipment is seismically tested in accordance with SPEC-11-00001-R and will not affect surrounding equipment during seismic events. Heat introduced by the equipment is reduced from that of the original equipment as the new equipment operates using low powered digital components whereas the old equipment replaced uses traditional high powered analog equipment.

The chiller controls are located in the equipment room used for no other purpose than for the Control Building Chilled Water System mechanical and controls equipment.

c. Have potential interaction between safety-related and non-safety related systems been addressed?

No new interfaces have been created as the result of this modification. All components installed by this modification are safety related. Existing interfaces with non-safety related equipment remain essentially unchanged. The existing non safety related first hit panel (IHA-PNL1) signals are double fused in isolation panel HVK-PNL2A(B) providing reliable isolation protection (Ref: ESK-06HVK14 sh. 2). Interfaces between the IHA-PNL1 and the chiller control panels HVK-PNL1A(B)(C)(D) will be removed by eliminating the cables between the chiller control panels and the isolation panels as the IHA-PNL1 indications are no longer needed and are provided by the new control panels HVK-PNL1A(B)(C)(D). Interfaces between safety related and non-safety related circuits in the Main Control Room are accomplished through existing isolation devices and existing fuses located in H13-PNL851(2) (Ref: ESK-10ANN29 Sh. 1). Both interfaces comply with the requirements of IEEE 384-1974 (Separation & Isolation) and IEEE 379-1977 (Single Failure).

The MCR alarm interface uses dry contacts (rated for the application)

provided by the new chiller controls to indicate a pre-trip condition. Power for the Alarm Circuits in the Main Control Room use non-safety power on the annunciator side and Class 1E power on the chiller control panel side of the isolation device maintaining separation requirements. No new fuses were required for the change as existing fuses were sized in accordance with external and unchanged MCR annunciator power requirements and IHA-PNL1 interfaces were eliminated.

d. Are the electrical loads associated with the upgraded system addressed in the design?

Yes. The control panel, hot gas bypass valve actuator, vane actuator, and associated skid mounted instrumentation will be powered by 120VAC. A review of the electrical design basis calculations indicates that adequate margin exists to accommodate the upgraded equipment. Calculation E-212 documents the 120VAC distributed load on panels supplying the Control Building Chiller control panels and are updated to reflect the new configuration. In addition, the factory acceptance testing for the new controls included verification that they will operate at the minimum supply voltage. (ref. 6216.210-085-015)

Upstream calculated load is of no consequence as these calculations (G13.18.3.6*016, E-132) model all 120/240 VAC panels as static loads and only compare minimum voltage at the panels to required voltage at the devices. In the case of calculation E-132, 120/240 VAC panels are not modeled at all; rather a demand factor of 1.00 was assumed to the running MCC load for full load and LOCA conditions. A demand factor of 0.50 was used for light load configurations. The MCCs are the source of the main feeds for the 120/240VAC distribution and are therefore bounded by the assumption of the E-132 calculation.

e. Does the plant HVAC have adequate capacity for the thermal loads of the upgraded system?

Yes. Calculation G13.18.2.1*059 describes chiller capacity and net reduction in capacity as the result of Control Building heat loads. Heat loads are tabulated in calculation E-225. Per calculation E-225, control panel heat loads for HVK-PNL1A (B)(C)(D) are not used directly. Rather their power sources, SCV-PNL8A1 (for A and C), SCV-PNL8B1 (for B and D), are used to determine watts generated as heat. The total watts generated is based on the panel rating (conservative) and not on actual loading (less conservative) therefore the change is bounded by existing calculated heat load generated.

f. Does the upgraded system meet the applicable requirements for separation, independence, and grounding?

Yes. Separation remains unaffected by the change. The system equipment is physically separated and protected with a barrier between divisions. Only skid mounted controls will be replaced. Interfaces with existing equipment external to the skid mounted equipment will remain unchanged as it relates to raceway, penetrations, and divisional separation. No new divisional interfaces are created as the result of this change.

Independence is unaffected by the change. Chilled water is supplied by two independent trains, either one of which is capable of meeting the total chilled water demand. Each train contains two 100-percent capacity electric motor-driven, centrifugal liquid chillers, two 100-percent capacity chilled water recirculation pumps, two 100-percent capacity condenser cooling water pumps, and one chilled water compression tank. One (1) chiller in each train is valved out and de-selected such that it will not start automatically and cannot be aligned from the Main Control Room. One chiller is in continuous operation and the remaining chiller (in the opposite train) is in standby awaiting an auto start signal should the operating chiller trip. The service water systems provide the chiller condenser cooling water. Each chilled water train, A and B, has separate connections to the corresponding service water train. Only skid mounted controls, instruments, actuators, and valves are being replaced. No new interfaces with independent trains are created by this change.

Grounding of the new panels will be performed in accordance with specification 248.000 and EE-033G which complies with existing design.

g. Does the upgraded system have adequate cabinet cooling?

Yes. The equipment has been qualified in the cabinet in which it is to be installed. The cabinet (panel) and associated instruments have been qualified for the environment in which they are to operate in accordance with SPEC-11-00001-R which uses River Bend Station environmental design criteria as delineated in 215.150.

h. Could a common cause failure result in a system-level failure based on the failure analysis?

No. Based on the qualification testing documented in qualification report 6216.210-085-010, the newly supplied control systems are reliable and there are no credible common cause failure modes.

As it relates to hardware (instruments, vane actuator, hot gas bypass valve / valve actuator, relays) the existing system contains four identical chillers with identical components for each skid. The new system will install similar components which are also identical from one skid to the next.

The new controls consist of a digital control panel, digital sensors for pressure and temperature, digital guide vane actuator, analog hot gas

bypass actuator, and analog CTs, PTs, and differential pressure transmitters.

- The new control panel contains a digital controller (HVK-PNL1A(B)(C)(D) LC-1A22) to monitor digital inputs (pressure, temperature, guide vane position) and provides digital commands to the guide vane actuator.
- The new controller also has a digital to analog converter installed in the new control panel which provides the interface for hot gas bypass control.
- CTs and PTs being analog interface with the digital controller via the digital starter module which accepts that CT / PT analog input. The starter module then provides this information digitally to the digital controller which both displays and issues control functions based on the input. The Potential Transformers are a new feature used for indication and capacity calculations. No trips are initiated as the result of the new PT feature.
- The differential pressure transmitters being analog have a digital to analog converter module installed in the control panel which allows communication with the digital controller. The controller displays the flow rates of service water and chilled water and uses these values for capacity calculations which are also displayed (all new features). No new trips are initiated as the result of the new DP features.

Given both old control system and new control system components installed use the same control system design from one skid to the next, there is no increase in the likelihood of a common mode failure being introduced above that which was previously installed. Diversity remains unchanged.

Furthermore, the new control panels are isolated digitally from each other and from network connections. The new control panels controllers are independent in system clock function. No synchronization occurs with the clock function and the clock functions are set independently. Furthermore, the controllers' internal clock utilizes a Century bit which is automatically set therefore issues related to Y2K have been eliminated.

As it relates to software and firmware, a defense in depth analysis (Attachment 1 to this 50.59) has been performed and concludes sufficient diversity exists to ensure common mode failures are not credible. Also, the

vendor supplied Failure Modes and Effects Analysis (Ref: N996-0108) provides a thorough evaluation of common mode failures and concludes no additional increase in likely hood of a system level failure. Given that each chiller's process parameters are slightly different, a common mode failure as the result of process parameter input is unlikely. Based on successful Seismic, Environmental, and EMI / RFI qualification testing as documented in qualification report 6216.210-085-010, it is not probable that a common mode failure exists as the result of external environmental influences. Given that V&V has been performed and the existing digital platform has been installed since 2008 in other plants and has been vetted of hidden software programming issues through use, it is not probable that logic errors could result in the failure of all four chillers simultaneously.

i. See subset below:

i. Is there reasonable assurance that the dependability of the system is sufficient (i.e. the likelihood of failure is significantly below that of single, active failures)?

A new single (or common mode) failure due to digital equipment has not been introduced that would result in the likelihood of a failure increasing. This is based on the fact that each chiller skid has a unique input footprint resulting in different system response to chiller input parameters. Per the Failure Modes and Effects Analysis for the new equipment, N996-0108 EC31808, "Given each chiller is distinct and separate, temperatures, pressures, currents, and voltages, will vary slightly from skid to skid presenting the digital chiller controls with a unique process finger print. As such, it is improbable that the twenty (20) inputs from one chiller to another will match identically creating a scenario where given a certain twenty (20) input configuration, the software or firmware fails. "

Also, Environmental influences (Seismic, temperature, humidity, EMI/RFI) have been eliminated as potential causes for failure through qualification testing as documented in 6216.210-085-010.

Finally, Software and firmware error risks have been reduced to that of standard off the shelf non digital components through the performance of verification and validation (ref, N996-0109), software quality assurance, failure modes and effects analysis, factory acceptance testing (ref. 6216.210-085-015), and performance vetting (using the platform or earlier versions of it in other plants since 2008) which has provided feedback and bug fixes to the initial offering.

ii. Was the application software developed under a 10 CFR 50 Appendix B, QA program using a documented life-cycle development process?

Although the chillers and their controls are nuclear safety-related, the replacement system is being procured as a commercial-grade

dedication item (CGI). It has been subjected to a CGI dedication process, under the auspices of the NQA program at a third-party qualifier. The selected third-party qualifier is NLI (Nuclear Logistics, Inc), which has CGI dedication experience, and has successfully qualified commercial grade chiller controllers for use in safety-related applications at nuclear power plants. NLI is listed on the qualified suppliers list for Entergy as an Appendix B program in accordance with EN-QV-122, "Qualified Suppliers List".

Software Lifecycle Management Plan was performed by NLI in accordance with the applicable recommendations of IEEE 1074-1995 and IEEE 1012-2004 per the NLI Verification and Validation Report N996-0109.

iii. Does the design comply with industry and regulatory standards?

Yes. The design was procured in accordance with SPEC-11-00001-R. The design was qualified in accordance with Qualification Plan 6216.210-085-009 and successful qualification is documented in Qualification Report 6216.210-085-010. This documentation states that the controls were Seismically Tested in accordance with IEEE 344-1975/2004, IEEE 323-2003, and the RBS specification SPEC-11-00001-R. Electromagnetic Interference/Radio Frequency Interference (EMI/RFI) qualification was performed in accordance with EPRI TR-102323, revision 3. Mild environment qualification by analysis and testing was performed in accordance with IEEE 323-2003. The Software V&V program was established in accordance with EPRI TR-106439 and IEEE 7-4.3.2-2003 Annexes C and D. The Software Lifecycle Management Plan was developed accordance with the applicable recommendations of IEEE 1074-1995 and IEEE 1012-2004. The Software Quality Assurance Plan was developed in accordance with the applicable requirements of IEEE 1012-2004 and the RBS specification. Human Factors Engineering (HFE) review of the human-system interface was performed in accordance with the applicable recommendations of IEEE 1023-1988 and NUREG 0700. A Requirements Traceability Matrix meeting the intent of IEEE 830-1998 was developed. A software configuration management plan was developed in accordance with the applicable requirements of IEEE 828-2005. Finally, a Failure Modes (ref. N996-0108, NLI FMEA) and Effects Analysis equivalent to the recommendations of IEEE 352-1987 was performed.

iv. Is there prior operating history for the digital device(s) and their firmware?

Yes. The digital devices including control panel modules, power

supplies, relays, vane actuator, hot gas bypass valve actuator, temperature, and pressure instruments have all had prior continuous use in safety related applications at McGuire (2008) and Catawba Nuclear (2009) Stations and have demonstrated dependability in these applications. These utilities have been consulted and indicate a greater degree of reliability (ability to stay online during degraded conditions and no controls related failures) and enhanced performance monitoring have been achieved. The Catawba and McGuire systems utilized the same sensors and control modules; however, the controller (Trane Tracer CH530) was an earlier version of the controller being used today (Trane Tracer AdaptiView using the UC800 controller).

The Trane Tracer CH530 control panels were offered from mid 2002 through 2008. These panels have many of the control features of new AdaptiView panels. However, they have limited communications and service interface capabilities. For improved user communications, logging and servicing capabilities, these panels are now being upgraded to new Trane Tracer AdaptiView control panels.

v. Has the platform been pre-qualified through NRC review?

No. However, this is not a requirement.

Per EPRI technical report TR-1001045 "Guideline on the Use of Pre-Qualified Digital Platforms for Safety and Non-Safety Applications in Nuclear Power Plants", "Although "pre-qualification" will significantly reduce the effort required for specific applications, most applications will require some additional work to address plant-specific and application-specific issues, and areas not fully covered by the "generic" qualification. Ultimately, acceptance of the digital installations will rely, at least in part, on engineering judgment."

vi. Does the design include features to detect, annunciate, and/or mitigate faults?

Yes. The new controller monitors chiller refrigerant temperatures, refrigerant pressures and electrical phase imbalances and adjusts chiller operation when conditions approach alarm limits. These new capabilities maximize the ability to keep the chiller running under conditions that would shut down the existing electromechanically controlled chillers. The existing controller only had static control capability which was based on a fixed refrigerant temperature. In the new design, if a limiting condition occurs, the Main Control Room will be notified via the Pre-Trip annunciator at which time Operations can observe locally what caused the limit. In the new design, should the chiller operate in limit mode, the real time capacity calculated value displayed at HVK-PNL1A(B)(C)(D) will alert Operations if insufficient capacity exists to satisfy accident condition capacity requirements of

158.98 Tons. (Ref: G13.18.2.1*059)

The new control system is capable of measuring heat exchanger approach tons, power consumption, power factor (uncorrected), compressor phase amps, and compressor phase voltage. The existing chiller controls did not provide monitoring capability of these parameters.

The new chiller controls has the capability to record data logs including ASHRAE 3 reports, custom reports, graphical custom historical data logs, purge reports, and maintains a 50 alarm log. The existing chiller controls required a separate panel (IHA-PNL1) known as the first hit panel to log events of the chiller. The existing monitoring capability of the IHA-PNL1 provided locally retrievable dry contact position indication on safety parameters associated with the existing Carrier chiller. This data was used strictly for post chiller trip analysis and provided no process monitoring or trend capability. The IHA-PNL1 data was not retrievable via SPDS. New control panel data will not be retrievable via SPDS and is only retrievable locally at the chiller control panel HVK-PNL1A(B)(C)(D).

Each of the systems, old and new, have the ability to alarm in the Main Control Room for problems at the panel locally alerting Operations. Main Control Room alarms consist of a Pre-Trip and Trip alarm. These two alarms remain unchanged however the new control system has the added benefit of monitoring sensor status (health) during standby conditions and annunciating the MCR pre-trip when such conditions occur.

vii. Has the system been tested under all normal and abnormal operating conditions?

Yes. The chiller controls and subcomponents are seismically qualified for a mild environment. They are subjected to EMI / RFI fields to ensure proper operation under specified conditions. The control

system and subcomponents are qualified for the environmental conditions (Temperature, humidity, radiation) where the equipment is to be located as specified in Environmental Design Criteria 215.150 and maintained updated in the EDC at River Bend Station. These conditions were specified during the purchase of the equipment in SPEC-11-00001-R. These conditions were translated into the qualification test plan 6216.210-085-009. Successful qualification testing was documented in 6216.210-085-009.

j. Is there a clear trend toward increasing the likelihood of malfunction of the SSC(s)?

No. The new controls upgrade functionally is a like for like replacement of the control panel and instruments with similar control panel and instruments with the exception of the Start to Start restriction changed for the new control system by EC-56609 which allows 2 starts within 20 minutes under Accident conditions. This function is automatically executed and requires no operator action. Under normal conditions, procedure administrative controls limit the number of starts within 20 minutes to 1. By allowing 2 starts within 20 minutes under Accident conditions, the probability of a complete loss of Control Building chillers is reduced. With the inclusion of digital controls comes additional complexity including software however this additional complexity is offset by the increased ability to monitor and regulate critical chiller process variables which increase the reliability of the chillers.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

The replacement of the chiller control system and subcomponents (instruments, valve / valve actuator, vane actuator) does not introduce the possibility of a change in the consequences of an accident because the Control Building chillers are not an initiator of any new accidents and no new failure modes are introduced. The Control Building chillers provide cooling to the Standby Switchgear rooms, the Control Building HVAC equipment rooms, and the Main Control Room. The cooling is required to mitigate the consequences of an accident by keeping

required equipment within design temperature ratings and by keeping Operations personnel in the Main Control Room comfortable.

The chiller control system design temperatures, flow rates, design pressures, and initiation logic remain unchanged. The new controls upgrade functionally is a like for like replacement of the control panel and instruments with similar control panel and instruments with the exception of the Start to Start restriction changed for the new control system by EC-56609 which allows 2 starts within 20 minutes under Accident conditions. This function is automatically executed and requires no operator action. Under normal conditions, procedure administrative controls limit the number of starts within 20 minutes to 1. By allowing 2 starts within 20 minutes under Accident conditions, the probability of a complete loss of Control Building chillers is reduced. Based on the fact that failure of the chillers to perform their design function is classified as unlikely based on one of four chillers being required to perform the design function, the design function to cool personnel and equipment will be satisfied. Because the design function is satisfied, the consequences of an accident remain unchanged. Dose consequences are unchanged. Therefore, the proposed change to replace the chiller control systems does not result in more than a minimal increase in the consequences of an accident as previously evaluated in the UFSAR.

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a. Does the system directly contribute to accident prevention or mitigation? If so, could the system cause the consequences (i.e. radiological release) of the accident to increase more than minimally?

No. The system does not directly contribute to accident prevention. Rather, the Control Building chillers provide support (cooling) to systems and personnel that are required to mitigate accident (reference Tech Spec Bases 3.7.3)

Support cooling is provided during accidents to maintain the environment within the bounds of the equipment's qualification. During accident conditions, equipment which is normally in standby starts generating heat in the Control Building and this heat requires removal to maintain equipment within temperature specification. Major heat sources include Control Building safety related equipment located in the Standby Switchgear Rooms (Inverters, Motor Control Centers, Switchgear), which are required to provide power to Safety Related equipment and instruments directly involved in the mitigation of accidents. Furthermore, the Control Room envelope is cooled by the chilled water provided by the Control Building Chillers via Control Building HVAC (HVC). The Control Building envelope is designed to maintain personnel conform and the safety related function of equipment given area temperatures of 80 degF for 30 days of continuous occupancy. The remainder of the Control Building including Standby

Switchgear Rooms and Chiller Rooms are designed to maintain the safety related function of the equipment given a 104 degF area temperature. (Ref: G13.18.2.1*018, G13.18.2.1*059, EDC)

Given the high reliability of the Control Building chillers (4 - 100% capacity chillers available) it is not probable that the support function will be rendered unavailable.

Note that one (1) chiller in each train is valved out and de-selected such that it will not start automatically and must be locally aligned in order to do so. One chiller is in continuous operation and the remaining chiller (in the opposite train) is in standby awaiting an auto start signal should the operating chiller trip.

b. Does the upgraded system exhibit a response time beyond current acceptance limits (e.g., because of sample period, increased filtering)?

No. While the upgraded system adds an additional 60 to 90 seconds for boot-up time if power was lost and restored (LOP), the Emergency Diesel Generator's load sequence on power restoration for the chillers is over 3 minutes. Therefore, the additional boot-up time is of no consequence and the upgraded system exhibits response times within current acceptance limits without any delays.

System response is similar between new and original systems with respect to chilled water temperature process adjustments. Starting and stopping response times for the compressor remain essentially unchanged except as noted below.

During normal operations, with panel power available, there exists a 51 second time delay in a chiller start logic that remains unchanged from old to new chiller controls. This 51 second time delay allows time for the safety string permissives to be met including oil pressure in the compressor to build up. In the existing design, the oil pump does not start immediately but is delayed for 23 seconds to allow for safety string logic to provide a permissive. No such delay is required in the new design therefor the oil pump pre-lubrication starts immediately upon receipt of a request for start (Ref: 0216.210-085-009, N996-0107).

During a loss and restoration of power event such a load shed and diesel re-sequencing during a LOP-LOCA, there is an added boot up time associated with the new panel of approximately 60 to 90 seconds. The boot up time required does not affect chiller operation under normal operating circumstances because the panels (one in service, one in standby) are normally energized. During a loss of power, the panels will de-energize as expected causing an immediate shutdown to the chiller. Once power is re-applied, the associated boot-up time of approximately 60 to 90 seconds occurs. Following the boot up time, a pre-lubricating time of 51 seconds is applied. Once pre-lubrication occurs, the compressor for the

associated chiller starts. Total time from power restoration to chiller available to start is approximately 2 to 2 ½ minutes.

An evaluation of the Diesel Loading Calculation, E-192, indicates that both oil pump and compressor are started within the same load block. Per diesel loading calculation E-192, either divisions chiller will sequence on in 211 seconds (3 ½ minutes). In addition, these the calculation shows that these loads occur in the 90 seconds to 10 minute load block which lumps the entire load at the 90 seconds mark (conservative) meaning the Diesel Calculation assumes everything from 90 seconds to 10 minutes starts at 90 seconds. It should be noted that the diesel loading calculation does not provide distinction that the lube oil starts before the compressor motor (conservative).

In the existing design there are cycle timers that are used to control the rate at which the associated chiller can assume load. The timer slows the rate of guide vane opening and prevents and rapid increase in compressor loading. The existing cycle timers allow the chiller to adjust for load demand every 1 second out of 4 seconds. In the new design, there is a softloading feature that serves an identical purpose to limit the load the chiller assumes to prevent rapid increase in compressor loading and thus reduce overshoot, rapid reduction, and possible chiller trip. There are 3 different settings used in the softload feature, each of which is adjustable from a minimal value to a maximum value. Two of the settings are time settings from 0 to 120 minutes and the third adjust compressor motor amps from 20% loaded to 100% loaded. The key advantage of the new system is that time is a variable setpoint whereas the older analog control system had a non-adjustable time ramp.

Although the change made by ECN 56609 will remove the interlock in the chiller logic associated with START to START wait time less than 20 minutes, procedural controls will prevent use of this relaxation, such that only under accident conditions will it be permissible for a chiller to start after having been started in the last 20 minutes. During accident conditions, it is imperative to maintain the safety function first (provide chilled water for control building HVAC thereby maintaining design basis temperatures) and demote motor protection to second tier consideration. By allowing the chiller to re-start if needed in less than 20 minutes from the previous attempt to start is a change from current design limitations. The new Start to Start limitation for Accident conditions is 2 starts within a 20 minute period.

Following modification, there will still be no credible scenario during which any chiller motor will be started in contravention of NEMA Standard MG 1

Section 1-20.43. This standard provides the framework for proper operation of induction motors used in the chiller application at River Bend as follows: Two starts in succession, coasting to rest between starts, with the motor initially at ambient temperature (Cold Start) or one start with the motor initially at a temperature not exceeding its rated load operating temperature (Hot start). NEMA Standard MG 1 is consistent with limitations described by the chiller motor manufactures' induction motor data (reference Spec 216.210) which states the motors are capable of 2 cold starts or 1 hot start. The chiller motor protection, though a second tier consideration, is therefore maintained following ECN-56609 modification.

Both existing and new designs provide a 150 second delay from Stop to Start after the first 20 minutes based on the fact that the potential to damage to the motor due to excessive starts has passed (inrush heating of the windings has cooled). The Stop to Start delay allows the chiller time to close the guide vane which eliminates starting the chiller under load. (Ref: 0216.210-085-009, 3216.210-085-001C page 75, N996-0107).

c. Does the system perform adequately under high duty cycle loading (e.g., computational burden during accident conditions)?

Yes. The chiller control systems, both old and new, do not experience computational burdens as the result of an accident. The existing control system did not contain digital controls therefore computation burden did not exist. The new control systems are dedicated to the purpose of controlling the associated chiller and the chiller's monitoring points do not change as the result of an accident. The same processes are monitored in normal and adverse conditions. There are no digital network connections therefore no opportunity to add computational burden as the result of increased data network traffic. Duty cycle remains continuous for the in-service chiller. Loss of power conditions result in shutdown and restarts are carried out in a pre-programmed fashion that is unchanging between normal and accident conditions.

d. Does the architecture of the system exhibit a single failure that results in more severe consequential effects (e.g., reduced segmentation due to combining previously separate functions, several input channels sharing an input board, central loop processor for many channels)?

The four (4) 100% capacity chillers and associated controls remain segmented and separate from the old to new design. No change in interlocks between chillers has occurred as the result of the upgrade.

For each skid, the new control panel replaces the old control panel. Internal to each control panel for the old design, discrete components existed that

are consolidated on the new design. The new design still has discrete input modules for incoming process parameters; however, these modules provide inputs to a single processor.

The lack of segmentation internal to the new panels is offset by the increased process monitoring, ability to self check internal circuits to determine validity of process data, ability to check sensor performance and determine if sensors are invalid, and ability to alter chiller performance to remain online during degraded conditions, whereas the older design (existing) would have tripped (e.g. condenser high pressure cutout). When operating in a degraded state, an alarm will provide indication both local and remote that such a condition exists allowing operations the ability to alternate chillers.

A single failure within a control system will at worst cause the associated chiller to trip. This worst case consequence, though less likely, is unchanged by this control system replacement.

e. Does the human-system interface design introduce increased burdens or constraints on the operators' ability to adequately respond to an accident, for operator actions credited in the licensing basis, such that there are more severe consequential effects (e.g., inability to access and operate more than one control at a time)?

No. Main Control Room controls and indications remain unchanged. Locally, the new design features a more robust human interface with a full color touch screen display that allows both quick assessment and more in-depth capabilities to diagnose chiller performance.

f. Could the new system create an environment (e.g., temperature, humidity, seismic, EMI/RFI emissions, airborne particulates) which adversely affects other equipment used for accident mitigation such that the consequences of an accident are more than minimally increased?

No. As stated previously, the new equipment is tested for both noise emissions and susceptibility in accordance with purchase specification SPEC-11-00001-R which invokes EPRI TR-102323, Guidelines for Electromagnetic Interference Testing in Power Plants. The new equipment is not an emitter of particulates and adds no humidity. The new equipment is seismically tested in accordance with SPEC-11-00001-R and will not affect surrounding equipment during seismic events. Heat introduced by the equipment is reduced from that of the original equipment as the new equipment operates using low powered digital components whereas the old equipment replaced uses traditional high powered analog equipment.

The chiller controls are located in the equipment room used for no other

purpose than for the Control Building Chilled Water System mechanical and controls equipment.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

The Control Building chiller system functions to maintain the environmental conditions in the control building in order to support Control Room habitability and operability of SSCs important to safety. The consequences of a SSC failure outside of the Control Building chiller system remain unchanged by this control system replacement.

Existing HVAC was evaluated in USAR Chapter 3.6 for effects of pipe breaks outside containment. USAR Chapter 3, Table 3.6A-24 lists HVAC as having been evaluated for such pipe breaks. There is no explicit listing for pipe breaks inside the HVAC Equipment rooms. A review of the rooms indicate the only large lines in the chiller rooms are Chilled Water and Service Water for use by the chillers. The pipe chase in Control Building is separate and does not affect the chillers. The only lines located in the chiller rooms are those that support chiller operation (HVK & SWP). Any rupture in a chiller room would likely cause the division of chiller with the rupture to become inoperable. Therefore, there is no increase in the Moderate or High Energy Line Break (MELB, HELB) analyses as a result of the addition of the new digital controls.

In the existing evaluation, the USAR 9.2.10.3 Failure Mode and Effects Analysis (FMEA) states that a loss of the control panel results in a loss of the associated chiller. The existing FMEA does not distinguish between the control panel and sensors and actuators on the skid. In the new digital controls upgrade, the replacement of these components does not alter the original failure mode (i.e. electronic controls fail resulting in loss of chiller). Dose consequences remain unchanged.

Therefore, Control Building chiller controls upgrade does not introduce the possibility of a change in the consequences of a malfunction or increase in dose because the Control Building chillers are not an initiator of any new malfunctions and no new failure modes are introduced.

NEI 01-01

a. Does the system play a role in mitigating the consequences (i.e. radiological release) of a malfunction? If so, would the change result in more than a minimal increase in the consequences of the malfunction?

No. The Control Building chiller system does not directly contribute to consequence mitigation. While the system functions to support operability of accident mitigating systems, the Control Building chillers do not contribute to increases in the consequences of a malfunction within these systems. Failure of the chillers results in the gradual increase in environmental temperatures allowing time for Operations to respond and achieve adequate cooling with the use of procedure AOP-0060 "Loss of Control Building Ventilation".

It is not probable that failure of the Control Building chiller could occur to the extent that failure of overall system would occur. Sufficient diversity exists in that each chiller can meet 100% of the design function which is to provide sufficient cooling capability to the Main Control Room, Standby Switchgear Rooms, and the HVAC Equipment Rooms. There are four (4) 100% capacity chillers grouped into two (2) independent trains. All four chillers would have to fail simultaneously which is improbable.

AOP-0060 states if Operations is unable to restart available HVK/HVC subsystem per SOP-0066 "Control Building HVAC Chilled Water System" use service water to cool the Control Building chilled water loops per SOP-0066 "Control Building HVAC Chilled Water System". AOP-0060 then directs operations to open key doorways and panels and provide temporary fans for ventilation in the switchgear rooms. The final step in this procedure is to restore a loop of chilled water.

b. Does the upgraded system exhibit the same failure modes affecting radiological releases as the system being replaced (e.g., fail low, fail high, fail-as-is, diagnostic failures)? If the failure mode is different, are the consequences increased beyond what was evaluated previously in the SAR?

Yes. The same failure modes exist for the new system as with the old system. As stated previously, in the existing FMEA referred to by USAR 9.2.10.3 but attached under a separate cover it states that a loss of the control panel results in a loss of that division of chiller. The existing FMEA does not distinguish between the control panel and sensors and actuators on the skid. In the new digital controls upgrade, the replacement of these components does not alter the original failure mode (i.e. electronic controls fail resulting in loss of chiller).

c. Is there a means available to alert the operators to the failure condition? Are the consequences bounded by other events evaluated in the SAR?

Yes. In the Main Control Room there exist two alarms, trip and pre-trip, that

alert operations to the malfunction or failure of a Control Building chiller. In addition, area temperatures are monitored which provide indirect indication to the design function of the chiller. If a chiller were to fail, the condition would be noted immediately.

Finally, local indication is enhanced with the new digital controls. The new controls are capable of measuring entering and leaving water, oil temperature, tons, power consumption, power factor (uncorrected), compressor phase amps, and compressor phase voltage. In addition, data logs include ASHRAE 3 reports, custom reports, graphical custom historical data logs, purge reports, and a 50 alarm log which are retrievable from the AdaptiView display screen. These features allow users to monitor and diagnose chiller operational trends which provide a clearer picture of chiller health increasing both reliability and availability. Self diagnostics of the chiller controls continuously monitor component health of the chiller control system for validity of signals and will indicate problems if detected.

Consequences of a chiller failure are bounded by the USAR in that there are 3 standby 100% capacity chillers available to replace the failed in-service or degraded chiller (total of 4 chillers, only 1 of which is required).

d. Can the system have an adverse impact on the installed environment (e.g., temperature, humidity, seismic, EMI/RFI emissions, airborne particulates) such that performance of an existing system used for accident mitigation will be more than minimally degraded compared to existing requirements?

No. As stated previously, the new equipment is tested for both noise emissions and susceptibility in accordance with purchase specification SPEC-11-00001-R which invokes EPRI TR-102323, Guidelines for Electromagnetic Interference Testing in Power Plants. The new equipment is not an emitter of particulates and adds no humidity. The new equipment is seismically tested in accordance with SPEC-11-00001-R and will not affect surrounding equipment during seismic events. Heat introduced by the equipment is reduced from that of the original equipment as the new equipment operates using low powered digital components whereas the old equipment replaced uses traditional high powered analog equipment.

The chiller controls are located in the equipment room used for no other purpose than for the Control Building Chilled Water System mechanical and

controls equipment.

5. Create a possibility for an accident of a different type than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

The Control Building chiller controls upgrade does not introduce the possibility of a new accident because the Control Building chillers are not an initiator of any accident and no new failure modes are introduced. Existing USAR analysis indicates the failure mode of the Control Building chiller controls results in the loss of the chiller. In the new design, failure of the Control Building chiller controls still results in the loss of the Control Building chiller. This single failure is previously analyzed as there are three remaining 100% capacity chillers that can satisfy the design function to provide cooling in the Control Building.

NEI 01-01

a. Have the assessments of system-level failure modes and effects for the new system or equipment identified any new types of system-level failure modes that could cause a different type of accident than presented in the plant SAR?

No. Failure of the new system controls still result in the loss of the entire chiller. In the existing evaluation the USAR 9.2.10.3 FMEA attached under a separate cover states that a loss of the control panel results in a loss of that division of chiller. The existing FMEA does not distinguish between the control panel and sensors and actuators on the skid. In the new digital controls upgrade, the replacement of these components does not alter the original failure mode (i.e. electronic controls fail resulting in loss of chiller).

b. Plant SAR analyses were based on credible failure modes of the existing equipment. Does the replacement system change the basis for the most limiting scenario?

No. While the USAR states that loss of ventilation and the chillers is not a credible event (Ref: USAR 3.11.4), it recognizes that such an event could occur as documented in USAR 9.2.10.2 which allows the use of Service Water to provide reduced capacity cooling should both trains of chilled water be lost. These actions are documented in AOP-0060 "Loss of Control Building Ventilation". This AOP states if Operations is unable to restart available HVK/HVC subsystem per SOP-0066 "Control Building HVAC Chilled Water System" use service water to cool the Control Building chilled water loops per SOP-0066 "Control Building HVAC Chilled Water System". AOP-0060 then directs operations to open key doorways and panels and provide temporary fans for ventilation in the switchgear rooms. The final step in this procedure it to restore a loop of chilled water. This scenario remains unchanged by this control system replacement modification.

c. Has power supply quality been considered (e.g., high harmonics from inverters, slow loss of voltage, or high voltage conditions)?

Power supply quality for the compressor motor is unchanged as the power supply source and motor remain unchanged. It should be noted in the old (existing) control system the power supply is not monitored for harmonics, slow loss of voltage, or high voltage conditions. Should such conditions occur in the old system, failure or shutdown would result. In the new system, power quality is monitored. This capability has been introduced by using potential transformers and current transformers. These new modules allow the new control panel to monitor power consumption, power factor (uncorrected), compressor phase amps, and compressor phase voltage. Furthermore, the new control system can compensate for the various conditions by limiting chiller capacity to reduce input power quality conditions to levels non-injurious to the equipment.

Power supply quality for the controls has been considered in the design of the new system. Qualification testing documented in Qualification Report 6216.210-085-010 and performed in accordance with Qualification Plan 6216.210-085-009 accounts for variations in supply voltage from108 Vac (90% of 120 Vac) to 132 Vac (110% of 120 Vac) which are the RBS anticipated outer limits for supply voltage. In addition, EMI / RFI testing was performed on the equipment to ensure proper operation and acceptable emissions.

d. Could the new system create an environment (e.g., temperature, humidity, seismic, EMI/RFI emissions, airborne particulates) which adversely affects other equipment and creates the possibility of an accident of a different type?

No. As stated previously, the new equipment is tested for both noise emissions and susceptibility in accordance with purchase specification SPEC-11-00001-R which invokes EPRI TR-102323, Guidelines for Electromagnetic Interference Testing in Power Plants. The new equipment is not an emitter of particulates and adds no humidity. The new equipment is seismically tested in accordance with SPEC-11-00001-R and will not affect surrounding equipment during seismic events. Heat introduced by the equipment is reduced from that of the original equipment as the new equipment operates using low powered digital components whereas the old equipment replaced uses traditional high powered analog equipment.

The chiller controls are located in the equipment room used for no other purpose than for the Control Building Chilled Water System mechanical and controls equipment.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

The malfunction of the Control Building chilled water controls is defined in the failure modes and affects analysis original to the USAR as a complete failure of any of the control panels resulting in total loss of the chiller. The original failure modes and effects analysis as listed in the existing USAR 9.2.10.3 for River Bend Station is high level and does not address components at the level of the digital control panel replacement. The existing USAR 9.2.10.3 FMEA states that a loss of the control panel results in a loss of that division of chiller. One (1) chiller in each train is valved out and de-selected such that it will not start automatically and must be locally aligned in order to do so. One chiller is in continuous operation and the remaining chiller (in the opposite train) is in standby awaiting an auto start signal should the operating chiller trip. For Division I, the loss of either Chiller A or C control panel (whichever is valved in) results in a loss of chilled water for Division I. For Division II, a loss of Chiller B or D control panel (whichever is valved in) results in a loss of chilled water for Division II. The original FMEA does not mention skid mounted instrumentation nor does it mention the Hot Gas Bypass Valve or Guide Vane actuators.

The replacement of the Control Building chilled water controls and instruments does not introduce the possibility for a malfunction of an SSC with a different result because the activity does not introduce a failure result different than that previously evaluated (total loss of control panel).

NEI 01-01

a. Does the change involve combining previously separate functions into one digital device such that a failure creates a result not bounded by the results of malfunctions previously considered in the UFSAR?

   No. While the digital upgrade does consolidate control functions under a single controller, the original design utilized a failure mode assessment with essentially the same result. While the original controller had separated functions into relays, controllers, and timers, these items were located in a single panel. As previously evaluated in the original Failure Modes and Effects Analysis (FMEA), the original evaluation lumps all controls under a single control panel; therefore, the consequences of the original FMEA are bounding. If a panel fails, the entire chiller fails.

b. Based on a qualitative assessment, is there reasonable assurance that failures due to software, including software common cause failures are unlikely (i.e. no more likely than other potential common cause failures such as maintenance or calibration errors that are not considered in the UFSAR)? If not, are the results of the software common cause failure different than (i.e. not bounded by) the results of the malfunctions considered in the UFSAR?

There is reasonable assurance that software related failures including common cause failures are no more likely to occur than those of corresponding analog components. Attachment 1 of this 50.59 evaluates common cause failures and concludes there is reasonable assurance that such failures are extremely low. In addition, this analysis concludes an alternate and diverse means is available in the interim (during chiller restoration) should such failure occur.

While the USAR states that loss of ventilation and the chillers is not a credible event (Ref: USAR 3.11.4), it recognizes that such an event could occur as documented in USAR 9.2.10.2 which allows the use of Service Water to provide reduced capacity cooling should both trains of chilled water be lost. These actions are documented in AOP-0060 "Loss of Control Building Ventilation". This AOP states if Operations is unable to restart available HVK/HVC subsystem per SOP-0066 "Control Building HVAC Chilled Water System" use service water to cool the Control Building chilled water loops per SOP-0066 "Control Building HVAC Chilled Water System". AOP-0060 then directs operations to open key doorways and panels and provide temporary fans for ventilation in the switchgear rooms. The final step in this procedure it to restore a loop of chilled water.

c. Could the environment in which the upgraded equipment operates cause a new type of failure (e.g., electromagnetic susceptibility in a higher frequency range)? Could the new system create an environment (e.g., temperature, humidity, seismic, EMI/RFI emissions, airborne particulates) which adversely affects other equipment and thereby creates the possibility of a different type of malfunction?

No. The new equipment is tested for both noise emissions and susceptibility in accordance with purchase specification SPEC-11-00001-R which invokes EPRI TR-102323, Guidelines for Electromagnetic Interference Testing in Power Plants. Per the qualification report, 6216.210-085-010 EC31808, "The guide vane actuator had radiated emissions higher than the allowables at 1 meter. Additional testing was performed at 1.8 meters and the emissions were lower than the allowables." The plant has verified via walkdowns and drawing review that there are no other digital components within 2 meters of the guide vane actuator other than the new NLI supplied digital equipment itself which was not susceptible to the emission. The report also states that the radiated emissions and susceptibility were acceptable with the display screen door open and that the testing was not performed with the main door

open. Based on this testing, the EMI/RFI qualification is acceptable and poses no risk to other plant equipment and will operate satisfactorily given existing EMI/RFI levels. It should be noted that during factory acceptance testing, no main panel door or other shielding was used and the equipment operated satisfactorily as documented in Factory Acceptance Test Report (FATR) 6216.210-085-015.

The new equipment is not an emitter of particulates (dust) and adds no humidity. The new equipment is seismically tested in accordance with SPEC-11-00001-R and will not affect surrounding equipment during seismic events. Heat introduced by the equipment is reduced from that of the original equipment as the new equipment operates using low powered digital components whereas the old equipment replaced uses traditional high powered analog equipment.

The chiller controls are located in the equipment room used for no other purpose than for the Control Building Chilled Water System mechanical and controls equipment.

d. Does the upgraded system have the same failure mode on loss of power as the system being replaced? If the failure mode is different, are the consequences increased beyond what was evaluated previously in the SAR?

On loss of power both old and new control systems shut down the chiller. No new consequences have been introduced with the installation of the new control system, however, the new chiller has the ability when power is restored to re-start even if it was started within the last 20 minutes. The older design will not allow for two starts in 20 minutes under loss of power conditions. The new design was changed by EC-56002 to allow for the new controls to start within 20 minutes of the last start following a loss of power to prevent a scenario in which Control Building chillers are out of service due to waiting on the Start to Start restriction of 20 minutes to clear.

Given that one chiller is running and the alternate divisions chiller is not running but in standby, during a loss of power event, it is presumed that the standby chiller would start if the running chiller was within its 20 minute start inhibit window. However, if the standby chiller was unavailable to start, the chiller that was running prior to the loss of power event would need to restart. By lifting the 1 start per 20 minute restriction, the chiller running prior to the loss of power event is allowed a second start within the 20 minutes thereby minimizing the possibility of all Control Building chillers being unavailable.

e. Is the response of the upgraded system on restoration of power different from that of the system being replaced? If so, are the consequences bounded by what was evaluated previously in the SAR?

Restoration of chiller control after power is lost remains unchanged. Chiller start control logic is provided external to the chiller controls so that the redundant system will start automatically. During loss of off-site power, the

pre-selected chiller compressor 1B or 1D starts up automatically in their proper standby bus loading sequences in Division II. In the event the Division II chiller fails to start automatically, the pre-selected chiller compressor 1A or 1C in Division I starts automatically after a time delay.

It should be noted that there is a boot-up time associated with the new controller that was not present on the existing controller. As discussed previously, during a loss and restoration of power event such a load shed and diesel re-sequencing during a LOP-LOCA, there is an added boot up time associated with the new panel of approximately 60 seconds. The boot up time required does not affect chiller operation under normal operating circumstances because the panels (one in service, one in standby) are normally energized. During a loss of power, the panels will de-energize as expected causing an immediate shutdown to the chiller. Once power is re-applied, the associated boot-up time of approximately 60 seconds occurs. Following the boot up time, a pre-lubricating time of 51 seconds is applied. Once pre-lubrication occurs, the compressor for the associated chiller starts. Total time from power restoration to chiller available to start is approximately 2 minutes.

An evaluation of the Diesel Loading Calculation, E-192, indicates that both oil pump and compressor are started within the same load block. Per diesel loading calculation E-192, either divisions chiller will sequence on in 211 seconds (3 ½ minutes). In addition, these the calculation shows that these loads occur in the 90 seconds to 10 minute load block which lumps the entire load at the 90 seconds mark (conservative) meaning the Diesel Calculation assumes everything from 90 seconds to 10 minutes starts at 90 seconds. It should be noted that the diesel loading calculation does not provide distinction that the lube oil starts before the compressor motor (conservative).

Internal time delays that start timing upon initial start of the diesel generator are present in both the existing chiller control and new system replacing it. If the new internal compressor start time delay were to fail such that no time delay occurred and an immediate compressor start was allowed, impact on the diesel loading would remain unchanged. The existing start time of the compressor is 211 seconds which includes the initial load sequencing delay for the diesel generator and the internal time delays needed for the compressor. This places the start of the compressor in the 1.5 to 10 minute load block. 211 seconds minus the 51 second pre-lube time delay is 160 seconds which is still within the 1.5 to 10 minute load block, therefore no impact to diesel loading would occur as the result of a time delay failure and the loading is enveloped by the existing calculation.

f. Does the system or equipment reset to operating parameters and settings established for the specific system, or does it go to a default set of parameters when the system is reset? If the system is reset with factory default parameters, what effect do they have on plant operation? Are the

consequences bounded by what was evaluated previously in the SAR?

There is no change in power loss and ability to retain settings from the old to new system conversion. The control system resets to operating parameters and settings established prior the power loss. The memory for the AdaptiView is nonvolatile type, so if power is lost, operating settings are retained. A life time battery is standard, which is used only to support the clock function for the chiller and failure of the battery does not result in loss of the settings. The change from old to new control system is therefore bounded by what was previously evaluated in the USAR.

g. Does the human-system interface (HMI) introduce failure modes different from those of the existing system? If so, are the results bounded by what was evaluated previously in the SAR?

There is no change to the Main Control Room controls or indications.

Locally, the human interface has changed and enhanced to provide better indication of chiller performance and health. A color touch screen monitor has been included in the digital upgrade to allow Operations to interface with the new control system.

Human factor analysis to ensure ease of use and understanding has been performed. Qualification report 6216.210-085-010 documents a human factors engineering (HFE) review of the human-system interface was performed in accordance with the applicable recommendations of IEEE 1023-1988 and NUREG 0700. Also an Entergy human factors review has been performed in accordance with EN-DC-163. In addition, vendor training has been included prior to implementation to ensure proper understanding. Finally, operation and maintenance procedural updates based on a benchmarking trip at McGuire have been performed.

It is therefore concluded that HMI failures have been reduced to that of the existing system and are bounded by what was previously evaluated in the USAR. Furthermore, given the color monitor with graphical user interface, better understanding of process parameters and chiller health is achieved resulting in improved HMI performance.

h. Have assessments of system-level failure modes and effects for the new system or equipment identified any new types of system-level failures (that are as likely to occur as those failure previously considered in the UFSAR) that would result in effects not bounded by the result previously considered in the SAR?

No. The new failure modes and effects analysis (ref. N996-0108, NLI FMEA) does not result in new system level failure. As previously stated, the failure mode is a complete loss of Control Building chiller which bounds the new Control Building chiller failure modes in their entirety.

7. Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The Control Building chiller control replacement does not affect a design basis limit of any fission product barrier as described in the USAR.

Fission product barriers are described in USAR:

- Chapter 4.2, Fuel and Cladding
- Chapter 5, Reactor Coolant System and Connected System
- Chapter 6.2, Containment System

The USAR, Chapter 1.2 defines those Nuclear Safety Systems and Engineered Safety Features required to maintain process parameters (water level, temperature, pressure) within design basis limits of the fission product barriers. It is the Class 1E equipment required to operate to maintain the design basis limit which rely on environmental conditions suitable for maintaining their own operating limits. The Control Building chillers provide the environment suitable for maintaining equipment and personnel in a condition required for the mitigation and maintenance of fission product barrier defense.

The Control Building chilled water controls upgrade does not result in a change in function or process (temperature of chilled water) that would cause any system parameter associated with a fission product barrier to be exceeded or altered nor does it result in system parameters associated with equipment required to mitigate and maintain those limits during accident conditions to be exceeded or altered. Therefore, the Control Building chiller controls upgrade does not result in a DBLFPB as described in the USAR being exceeded or altered.

NEI 01-01

a. Are any of the numerical values in the UFSAR that are used directly in the determination of the integrity of the fission product barriers associated with the change? Would the digital upgrade result in any of these values being exceeded or altered?

   No. Upgrading the chiller controls and instruments do not affect numerical values associated with fission product barriers. All numerical changes are the result of new instrument additions or deletions whose only impact is on the control of the chilled water for the Control Building. Electrical panel loading is reduced and is therefore conservative. The temperature setpoints

of the Main Control Room or Control Building will not change as the result of the upgrade.

b. Has the digital upgrade decreased the channel trip accuracy beyond the acceptance limit?

No. The accuracy of protective trip features internal to the chiller logic is evaluated under EC-31803 and found acceptable. External trips remain unchanged as the result of this modification. In addition, this modification has no impact on any trip setpoints, channel response times, or indication accuracies associated with any SSC other than the Control Building Chillers.

c. Has the digital upgrade increased the channel response and/or processing time beyond the acceptance limit?

Both the ability to process information and respond to the information has been enhanced as the result of this change. Converting from analog (mechanical movement) to digital (electronic) signals will reduce time delays from process sample to controller response.

Additionally, the ability to control the speed at which the chiller assumes load will maintain existing times or enhance existing times through the use of an adjustable time delay that did not exist prior to the change.

In addition, this modification has no impact on any trip setpoints, channel response times, or indication accuracies associated with any SSC other than the Control Building Chillers.

d. Has the digital upgrade decreased the channel indicated accuracy?

No. The chiller controls upgrade does not replace the Control Building temperature indicators which indicate ambient conditions in the Main Control Room and Control Building therefore these accuracies remain unchanged.

Skid mounted indications do not affect fission product barriers. Accuracies of the instrumentation are of no consequence to fission product barriers.

In addition, this modification has no impact on any trip setpoints, channel response times, or indication accuracies associated with any SSC other than the Control Building Chillers.

8. Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

As described in NEI 96-07, Revision 1, a method of evaluation is the calculation framework used to evaluate behavior or response of the facility or Structure, System, or Component (SSC). Those subject to 10 CFR 50.59 are methods of evaluation used in the analysis that demonstrate that design basis limits of fission product barriers are met or those used in the USAR safety analysis including

containment, ECCS and accident analysis presented in USAR Chapter 6 "Engineered Safety Features" and USAR Chapter 15, "Accident Analysis". NEI 01-01, Revision 1 guidance and NEI 96-07 are both considered in the evaluation.

The Control Building Chilled Water System is not credited with nor is it used in the method of analysis to determine design basis limits of fission product barriers; therefore, replacement of the chiller controls, instruments, and certain valves and auxiliary equipment does not affect the method of evaluation for design basis limits of fission product barriers previously evaluated in the USAR.

The Control Building Chilled Water System is credited with supporting Emergency Core Cooling System functions as described in USAR Chapter 6. Primarily, Chapter 6 describes evaluations which ensure the Main Control Room (MCR) Habitability System remains capable of performing its function. MCR habitability functions are to ensure that the Main Control Room remains comfortable and ensure that Control Building equipment is maintained within the temperature and humidity requirements for which it is rated. These two requirements, personnel comfort and equipment within temperature and humidity ratings, are required prior to, during, and post accident for 30 days. The evaluation concludes that these functions will be performed given a single active component failure coincident with a loss of offsite power. This is ensured by the requirement that the equipment be redundant, Seismic Category I, powered from Class 1E buses, and Safety Class 3 (Ref: USAR 6.4.1). Design and operating codes are listed in USAR 6.4.5.2 and include applicable ASME, ANSI, IEEE, and NEMA standards.

The modification replaces aging equipment with new equipment similar to a like for like replacement. The modification does not change the redundancy scheme, does not alter initiation or tripping schemes, does not change process parameters, and does not change design basis functions. One enhancement provided by EC-56609 is the ability to start twice within a 20 minute period under Accident conditions which minimizes the likelihood of being without a chiller while waiting for the 20 minute Start to Start time to elapse. This function is automatically executed and requires no operator action. Under normal conditions, procedure administrative controls limit the number of starts within 20 minutes to 1. By allowing 2 starts within 20 minutes under Accident conditions, the probability of a complete loss of Control Building chillers is reduced.

The control panels, associated instrumentation, valves, and valve/vane actuators used to replace existing equipment are tested in accordance with the same methodologies as that of the original equipment. As specified in purchase specification SPEC-11-00001-R, the equipment is supplied as Class 1E and is seismically qualified for Category I service in accordance with IEEE 323-1974 and IEEE 344-1974. Pressure boundaries comply with ASME Boiler and Pressure Vessel Code, Section III, Class 3. Guide vane and hot gas valve replacements comply with ANSI B31.5 and B31.5a, Code for Pressure Piping. Motors are not being affected by the change therefore NEMA MG-1 criteria remain unaffected.

NEI 01-01

a. Does the upgrade involve a change to any element of the analytical methods that are described in the UFSAR which are used to demonstrate the design meets the design basis or that the safety analysis is acceptable?

   No. The method used in the USAR dictates full redundancy such that a single failure will not prevent the chillers from performing their design basis function. The method also dictates that a single chiller is sufficient to provide for all cooling needs previous to, during, and following design basis event. Finally, the method dictates equipment will be qualified to the various codes and standards listed above. Independence, redundancy, capacity, and qualifying standards remain unchanged.

b. Does the change involve use of a method of evaluation not already approve by the NRC?

   No. The method of evaluation used, independence (redundancy, capacity, and qualifications) remain unchanged from old design to new design.

Therefore, the change 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.

DEFENSE IN DEPTH ANALYSIS

1. PURPOSE

The NRC's approach to the review of digital systems focuses, to a large extent, on confirming that a disciplined process was applied in developing the system. The NRC's position on quality of software for use in safety systems is stated in the Standard Review Plan Chapter 7, Branch Technical Position (BTP) HICB-14, "Guidance on Software Reviews for Digital Computer Based Instrumentation and Control Systems".

NRC considers that there is some level of residual susceptibility to common mode software failure when redundant channels of a safety system use the same software. For systems of the highest safety significance, namely the reactor trip system (RTS) and engineered safety features actuation system (ESFAS), NRC expects the licensee to address the potential for software common mode failure through a Defense-in-Depth and Diversity (D-in-D&D) evaluation.

The NRC staff has established a methodology and acceptance criteria for Defense-in-Depth and Diversity evaluations. The methodology and acceptance criteria are documented in Branch Technical Position HICB-19. (Ref: EPRI TR-1001045, "Guideline on the Use of Pre-Qualified Digital Platforms for Safety and Non-Safety Applications in Nuclear Power Plants)

NRC established the following position on D-in-D&D for modifications to operating plants:

a. The applicant/licensee should assess the defense-in-depth and diversity of the proposed instrumentation and control system to demonstrate that vulnerabilities to common-mode failures have been adequately addressed.

b. In performing the assessment, the vendor or applicant/licensee shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) using best-estimate methods. The vendor or applicant/licensee shall demonstrate adequate diversity within the design for each of these events.

c. If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, should be required to perform either the same function or a different function. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions.

2. Four Lines Of Defense Criteria

Defense-in-depth is the concept of multiple lines of defense against a perceived threat so that if one line of defense is penetrated, another line is invoked to limit the damage caused by the penetration. This can be carried through several levels. Per the Nuclear Regulatory Commission (NRC) Branch Technical Position HICB-19 there are four echelons of defense against common mode failure. These are:

a. Control System: The control echelon consists of that non-safety equipment which routinely prevents reactor excursions toward unsafe regimes of operation, and is used for normal operation of the reactor.

   Reactor controls are located in the Main Control Room. The Control Building chillers are relied upon to provide ambient temperatures in the Main Control Room for the habitability of operations personnel and to keep equipment required for the control and protection of the reactor within their design temperature ratings. In the unlikely event of loss of Main Control Room air conditioning, the operators have time to utilize AOP-0060 "Loss of Control Building Ventilation" and if necessary, a plant shutdown can be performed from the remote shutdown panel as discussed in USAR Section 7.4.

b. Reactor Trip System (RTS): The reactor trip echelon consists of that safety equipment designed to reduce reactivity rapidly in response to an uncontrolled excursion.

   At River Bend Station, the RTS system is known as the Reactor Protection System (RPS). RPS initiates a rapid, automatic shutdown (scram) of the reactor. It acts in time to prevent fuel cladding damage and any nuclear system process barrier damage following abnormal operational transients. The reactor protection system overrides all operator actions and process controls and is based on a fail-safe design philosophy that allows appropriate protective action even if a single failure occurs. (Ref: USAR 1.2.2.4.1)

   As it relates the chiller controls upgrade, RPS is powered from non-safety related motor-generator sets which are cooled by HVAC using Control Building chilled water. The RPS system is fail-safe such that a loss of RPS power results in an automatic reactor shutdown.

c. Emergency Safety Feature Actuation System (ESFAS): The ESFAS echelon consists of that safety equipment which removes heat or otherwise assists in maintaining the integrity of the three physical barriers to radioactive release (cladding, vessel, and containment).

   At River Bend Station, the ESFAS system is known as Emergency Core Cooling Systems (ECCS) and Engineered Safety Features (ESF). USAR 1.2.2.4 lists twenty (20) systems that make up the diverse group.

One set of Engineered Safety Features (ESFs) are the habitability systems. The Control Building chillers are one of the habitability systems and are responsible for cooling equipment in the Control Buildings Main Control Room, Standby Switchgear rooms, and HVAC equipment room. (Ref: 1.2.2.4.20)

d. Monitoring and Indicators: The monitoring and indication echelon consists of sensors, displays, data communication systems, and manual controls required for operators to respond to reactor events.

Monitoring and indications in the Main Control Room are accessible by Operations and these devices are cooled by HVAC which uses the Control Building Chilled Water System.

As it relates to the replacement of the chiller controls, manual control switches are provided in the Main Control Room for operation of the chiller compressors (HVK-CHL 1A, B, C, D). Control logic is provided to prevent starting a compressor until normal chilled water flow through the evaporator and normal service water through the condenser have been established. An extreme low flow condition stops the chiller compressor. Control logic is provided so that the redundant system will start automatically. During loss of off-site power, the pre-selected chiller compressor 1B or 1D starts up automatically in their proper standby bus loading sequences in Division II. In the event the Division II chiller fails to start automatically, the pre-selected chiller compressor 1A or 1C in Division I starts automatically after a time delay.

A chiller automatic trip alarm is provided in the Main Control Room. A chiller compressor pretrip alarm is also provided in the Main Control Room. A data acquisition system is provided to monitor the status of safety controls during chiller compressor startup, operation, and trips.

With the exception of the data acquisition system, all other features as noted above remain unchanged. Data acquisition points will be modified to reduce the number of points available. These points will be transferred to the new control panels which are capable of displaying the existing data points in addition to new data points making the data acquisition panels data points un-necessary.

3. Function of Control Building Chilled Water System

   The Control Building Chilled Water System is considered an auxiliary system and is not directly involved in mitigating the consequences of an accident. It is not part of the Reactor Protection System (RPS). It is an integral feature however in the Engineered Safety Feature (ESF) to cool the environment in Control Room to provide comfort for Operations and limit equipment temperature to rated levels. The system is designed to provide chilled water to the cooling coils in the air supply ventilation systems for the main control, Standby Switchgear, and Chiller Equipment rooms during all modes of plant operation, including DBA conditions. (Ref: USAR 1.2.2.8)

4. Loss of Control Building Chilled Water System

   Total loss of Control Building chilled water is not considered a credible event. USAR 3.11.4 evaluates if a loss of Control Building ventilation and subsequently the loss of Control Building chilled water is possible. The USAR evaluation concludes the following design features preclude the possibility of a total system failure for the Control Building ventilation system serving the main control, Standby Switchgear, and Chiller Equipment rooms where equipment is required to function during and following a DBA:

   a. All HVAC systems serving these equipment areas are designed to Seismic Category I requirements.

      The chiller control upgrade has been qualified to Seismic Category I requirements as specified in purchase specification SPEC-11-00001-R and documented in qualification report 6216.210-085-010.

   b. Sufficient redundancy in equipment and power supplies is provided so that no single active component failure can result in loss of HVAC system function.

      Single active component failures analysis is discussed in USAR section 15.0.3.2.1.3. The chiller control upgrade does not alter the redundancy in equipment and power supplies. The Control Building Chilled Water System consists of two redundant, closed loop chilled water trains. Each Control Building chilled water train consists of two 100-percent capacity, mechanical refrigeration water chillers, two chilled water circulation pumps, and associated piping, valves, and instrumentation.

      In the event both chilled water trains fail, partial cooling can be achieved by using the Standby Service Water instead of the chilled water as described in USAR Section 9.2.10.3. In the unlikely event of loss of Main Control Room air conditioning, plant shutdown can be performed from the remote shutdown panel as discussed in USAR Section 7.4.

      With the upgrade of the Control Building chiller controls, these conclusions remain unchanged.

c. Redundant HVAC systems are connected to separate and independent onsite standby power supplies to assure system operation upon loss of offsite power.

There are four trains of chiller controls. These controls are grouped into two per separated division. The Class 1E power systems are designed as a two Division system, with either Division 1 or 2 being adequate to safely shutdown the unit. Division 1 and 2 are the independent divisions which power the Control Building chilled water chillers. Division 1 powers HVK-CHL1A and HVK-CHL1C. Division 2 powers HVK-CHL1B and HVK-CHL1D. (Ref: USAR 1.2.1.2.7.3)

Each division is backed up by a diesel generator which provides the power to the division upon a loss of offsite power. The chiller control upgrade will utilize the same divisionally separated power as that of the existing chiller controls. In addition, the amount of power consumed will be reduced resulting in a reduction of load on each divisional power supply. Diesel generators will remain unchanged.

d. Failure modes for isolation valves and dampers are described in Section 9.4 of the USAR. Valves or dampers required for operation after postulated accidents fail in the safe position.

The chiller control upgrade does not alter isolation valves or dampers therefore failure modes for isolations valves and dampers will remain unchanged after the chiller control upgrade.

e. Equipment outside the containment building required to operate following a LOCA or a high-energy pipe break is so located that it is not exposed to resultant post accident ambient conditions or is designed to withstand these severe conditions.

The chiller control upgrade does not alter the location of the Control Building chilled water controls which are located in the HVAC rooms on the 98' elevation of the Control Building. In addition, existing power supplies are located in the Standby Switchgear rooms also located on the 98' elevation of the Control Building. Per the existing environmental design criteria (215.150), the HVAC and Standby Switchgear rooms are a mild environment and are not exposed to LOCA or a high-energy pipe break resultant post accident ambient conditions.

f. Instrumentation and controls which incorporated audible and visual alarms enable the operator to continuously monitor the HVAC systems' performances. In the event of system malfunction, the operator has the capability to switch manually to the HVAC standby equipment.

Controls and instrumentation in the Main Control Room remain unchanged. The new control system locally will change however the change enhances monitoring and alarm capability. The new controls are capable of measuring entering and leaving water, oil temperature, tons, power consumption, power factor (uncorrected), compressor phase amps, and compressor phase voltage. In

addition, data logs include ASHRAE 3 reports, custom reports, graphical custom historical data logs, purge reports, and a 50 alarm log which are retrievable from the AdaptiView display screen. These features allow users to monitor and diagnose chiller operation trends.

Conclusion - Based on the above features and the detailed HVAC systems' evaluation in USAR Section 9.4, only partial loss of the ventilation or air-conditioning system could occur in areas where equipment required to function during and following a DBA is located. This loss would not adversely affect the availability of the safety related equipment to function during and following a DBA. Equipment is qualified for the limiting environmental service conditions for which it must function assuming loss of non-Class 1E HVAC systems. Common Mode Failure is address separately. See item 5 below.

5. Common Mode Failure (ref. N996-0108, NLI FMEA)

The above evaluation (Loss of Control Building Chilled Water System) does not consider the possibility of a common mode failure due to software related errors which would result in a complete loss of chilled water thus changing the analysis of the USAR. As stated previously, common mode failures are a concern. These failures can result from external sources (environmental, seismic, inputs) or internal sources (software algorithms).

a. External Sources

<u>Environmental</u>

Environmental influences on software / firmware can occur as the result of temperature, radiation, humidity, and EMI / RFI. These external influences can impact the way digital electronics behave. As such, extensive qualification testing has been performed using River Bend Station specific environmental parameters. These parameter requirements were invoked by specification SPEC-11-00001-R and successful qualification documented in report 6216.210-085-010.

In addition, there are two (2) independent divisions of chillers housed in two (2) physically separated rooms. Given the physical separation, it is unlikely that each chiller control skid will experience the exact same environmental influences. Given these environmental influences are not identical, it is logical to conclude equipment response to the environmental influences will not be identical.

It is therefore concluded that there are no increases in common mode failures associated with environmental conditions.

Seismic

Seismic activity and its affect on software / firmware can occur as the result of poor connections from semiconductor processes. These external influences can impact the way digital electronics behave. As such, extensive qualification testing has been performed using River Bend Station specific seismic response spectra. These seismic criteria were invoked by specification SPEC-11-00001-R and successful qualification was documented in report 6216.210-085-010.

It is therefore concluded that there are no increases in common mode failures associated with seismic conditions.

Data Inputs

Diversity in the existing control system exists to the point that each of the four (4) chiller control skids has multiple components which monitor various processes. These monitored processes provide the means to control and alert operations to the status of the chiller. The instrumentation monitoring the chiller processes and status also provide the permissive to start and trip the respective chiller. Each of these processes instrumentation is replicated for each of the four (4) chiller control systems. Each component albeit different with respect to its function and on a specific skid is the same across the four (4) skids.

With the implementation of the new digital control upgrade, diversity is maintained in that each of the four (4) chiller control skids will maintain multiple components which monitor various processes. In the new design, the processes and components used to monitor them will increase thus creating more diversity. As with the existing design, the new design will maintain a replication of the design across all four (4) chiller control skids thus no reduction of diversity will occur from one skid to another. Given that both existing Carrier and new AdaptiView controls are safety related, each has gone through a rigorous quality assurance process in accordance with 10CFR50 Appendix B. Both the Carrier controls and the NLI supplied AdaptiView controls have 10CFR50 Part 21 reporting requirements such that any defects discovered are reported. Based on these sets of facts and similarities, no increase in hardware common mode failure rate has been introduced as the result of the component replacements. The risk of common mode failure due to the inputs is negligible as the result of the unique and independent process variable inputs each chiller control system monitors. While the chillers are considered identical, processes (flow, temperature, pressure, start, and stop inputs) are generated independently and present a unique set of parameters at any given time on any give chiller control input.

An operator of the equipment providing input erroneously is another form of input that can be a common mode failure. Given this type of error is just as likely on non-software based systems, it was not considered in the evaluation. Human factors to ensure ease of use and understanding has been performed.

Qualification report 6216.210-085-010 documents a human factors engineering (HFE) review of the human-system interface was performed accordance with the applicable recommendations of IEEE 1023-1988 and NUREG 0700. Also an Entergy human factors review has been performed in accordance with EN-DC-163. In addition, vendor training has been included prior to implementation to ensure proper understanding. Finally, operation and maintenance procedural updates based on a benchmarking trip at McGuire have been performed.

It is therefore concluded that there are no increases in common mode failures associated with inputs.

b. Internal Sources
Internal sources are inherent software and firmware errors that result in a failure to start the chiller, failure to stop the chiller, or failure to regulate chilled water temperature. While internal sources of failure can not be eliminated, they can be reduced to levels improbable through testing, evaluation, and quality assurance.

Qualification and Factory Acceptance Testing

Qualification testing of the equipment has been successfully completed and documented in Qualification Report 6216.210-085-010. Testing was performed in accordance with Qualification Plan 6216.210-085-009. This testing included seismic, EMI/RFI, Factory Acceptance Testing (ref. 6216.210-085-015), Demonstration Testing, and Closed Loop Simulation testing.

As part of the qualification plan, Factory Acceptance Testing (FAT) and dedication testing of the equipment (ref. 6216.210-085-015) was successfully performed in accordance with Purchase Specification SPEC-11-00001-R. As part of this testing, a demonstration test which is a representative mock up of the chiller, instruments, and controls is assembled. The demonstration test verifies proper operation of the controller over the duration of seventy-two (72) hours.

For those aspects of testing that were destructive, closed loop simulation testing was performed. Simulation testing also includes a first-order model of dynamic process behavior, in order to obtain preliminary tuning parameter values with minimal overshoot.

Finally, seismic testing and EMI/RFI testing were also performed as already mentioned.

Quality Assurance (Ref. N996-0109, NLI V&V)

Software Lifecycle Management Plan was performed in accordance with the applicable recommendations of IEEE 1074-1995 and IEEE 1012-2004. A Software Quality Assurance Plan was developed in accordance with the applicable requirements of IEEE 1012-2004 and the RBS specification SPEC-11-00001-R. The Requirements Traceability Matrix meets the intent of IEEE 830-1998. The software configuration management plan was implemented in

accordance with the applicable requirements of IEEE 828-2005. Based on adherence to these standards, the software is of high quality thus reducing the likelihood of failure.

Evaluation (Ref. N996-0108 NLI FMEA, N996-0109 NLI V&V)

Run time in the industry is another method of verifying software / firmware are error free. By reviewing AdaptiView control performance at previously installed locations, reliability of the software / firmware can be established using hard data. Real world applications where the environments and conditions are uncontrolled are some of the most reliable inputs into a determination on reliability and thus reduced risk of software / firmware induced common cause failures.

NLI conducted a Commercial Grade Survey of Trane and documented the number of AdaptiView units installed. The current processor software version was implemented into production in May 2009. The Tracer UC800 controller was launched during 2008. The operating experience information is documented in the software dedication/V&V report (ref. N996-0109, NLI V&V Report).

There are several nuclear facilities using the CH530/CH531 controller which utilizes the same LLID and previous version software. For purposes of comparison these plants were consulted for reliability. Duke McGuire and Catawba installed and are operating the digital NLI supplied Trane controllers. Four units have been installed with a total operating time of approximately 12 operating years.

One issue reported was the lockup of the touchscreen on one control panel. The unit reset properly when power was removed and restored. The unit was returned for testing at NLI and Trane. Neither NLI nor Trane could reproduce the problem. Trane provided feedback that they had not seen this issue on any commercial units.

At the Nuclear HVAC Utility Group meeting in July 2008, several nuclear facilities discussed their successes with digital controls upgrades. One facility installed the Trane digital controls systems and went from monthly chiller trips to 54 months with zero trips. A new design to replace the entire control system for the essential chillers with a digital upgrade to increase reliability and availability of the chillers should be considered for Vogtle. (Ref: EPIX Vogtle Unit 2 - Failure Number : 472, 06/04/2008)

Another supporting argument in the analysis of a common cause failure as the result of a software / firmware deficiency is the staged multi-year installation of the upgraded AdaptiView controls. RBS will install one AdaptiView chiller per year for the next four years. This will allow sufficient time to gradually gain confidence in the systems software / firmware before deploying across Divisions.

- Year 1 – Division II (HVK-CHL1D) installed
- Year 2 – Division I (HVK-CHL1A or C) installed
- Year 3 – Division II (HVK-CHL1B) installed
- Year 4 – Division I (HVK-CHL1A or C) installed

6. Evaluation (Branch Technical Position HICB-19)

The Control Building Chillers are required for all events evaluated at River Bend Station. This is based on the fact that cooling provided by the chillers is used in the Main Control room for personnel comfort and equipment temperature ratings in the Main Control Room, Standby Switchgear, and Chiller Equipment rooms.

Per USAR Table 3.11-2, HVK is required for the following accident analysis events:

- Steam Line Break Outside Containment
- Feedwater Line Break Outside Containment
- LOCA
- High Energy Line Break Outside Containment (RWCU/RCIC)
- Rod Drop
- Fuel Handling
- Recirculation Pump Seizure
- Recirculation Pump Shaft Break
- Main Condenser Gas Treatment System Failure

Per the USAR, Chapter 15, Transient and Accident Events requiring Control Building Chilled Water (HVK) are:

- Decrease in Core Coolant Temperature
- Increase in Reactor Pressure
- Decrease in Reactor Coolant System Flow Rate
- Reactivity and Power Distribution Anomalies
- Increase in Reactor Coolant Inventory
- Decrease in Reactor Coolant Inventory
- Radioactive Release from a Subsystem or Component
- Anticipated Transient Without Scram

Each of these scenarios assumes that ventilation and subsequently HVK is operating and maintaining the environment. With the loss of HVK, chilled water becomes unavailable and thus ventilation is lost. Loss of ventilation does not result in sudden failure of the equipment cooled by HVK. Neither does loss of ventilation result in immediate un-inhabitability of the Control Room. Rather, heat being removed ceases and a gradual heat up of the affected areas commences. All functions remain available until the 104 degree F rating is exceeded. There is ability to slow this process of heat up as described below which allows time for ventilation to be restored.

Sufficient diversity exists in that each chiller can meet 100% of the design function which is to provide sufficient cooling capability to the Main Control Room, Standby Switchgear Rooms, and the HVAC Equipment Rooms. There are four (4) 100% capacity chillers grouped into two (2) independent trains.

Each chiller has various input parameters that are independent of the other chillers parameters making inputs completely independent and diverse. No two input parameters will be identical. This provides a diversity in that input sets are unique to each chiller.

Internally, control systems and software are identical from one chiller control system to the next however testing, operating experience, factory acceptance testing (ref. 6216.210-085-015), and verification and validation (ref, N996-0109) have reduced software errors resulting in failure to a level equivalent to the failure probabilities of the controls being replaced. Therefore, an increase in likelihood of common mode failure is negligible.

Finally, should a common mode failure occur, there are diverse means of providing cooling to the Control Building areas at reduced capacity. These actions are documented in AOP-0060 "Loss of Control Building Ventilation". This AOP states if Operations is unable to restart available HVK/HVC subsystem per SOP-0066 "Control Building HVAC Chilled Water System" use service water to cool the Control Building chilled water loops per SOP-0066 "Control Building HVAC Chilled Water System". AOP-0060 then directs operations to open key doorways and panels and provide temporary fans for ventilation in the switchgear rooms. The final step in this procedure is to restore a loop of chilled water.

As discussed above, the Control building Chiller system can neither cause, prevent or directly mitigate a breach of any fission product barrier. Based on these facts, the changes proposed do not result in radiation release exceeding 10% of the 10 CFR 100 guideline value.

Below is a list of acronyms used in this evaluation

Acronym	Meaning
ANSI	American National Standards Institute
ASHRAE	American Society of Heating, Refrigeration, and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulation
CHL	Chiller
CT	Instrument Current Transformer
DBA	Design Basis Accident
DP	Differential Pressure
EB	Ductwork Drawings
EC	Engineering Change
ECCS	Emergency Core Cooling System
EDC	Environmental Design Criteria
EE	Electrical Drawings
EHS	480V Electrical Distribution (Class 1E MCC)
EJS	480V Electrical Distribution (Class 1E SWG)
EMI	Electromagnetic Interference
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
ESK	Elementary Schematic Diagram
FAT	Factory Acceptance Testing

Acronym	Meaning
FMEA	Failure Modes and Effects Analysis
HFE	Human Factors Engineering
MCC	Motor Control Center
MCR	Main Control Room
MOV	Motor Operated Valve
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturers Association
NLI	Nuclear Logistics Incorporated
NRC	Nuclear Regulatory Commission
HVAC	Heating, Ventilation, & Air Conditioning
HVC	HVAC – Control Building
HVK	Control Building Chillers
IEEE	Institute of Electrical and Electronics
IHA	Information Handling System
LLID	Low Level Intelligent Device
LOCA	Loss of Coolant Accident
LOP	Loss of Offsite Power
LSK	Logic Diagram Drawing
NQA	Nuclear Quality Assurance
PDS	Differential Pressure Switch
PDI	Differential Pressure Indicator
PS	Pressure Switch

Acronym	Meaning
PID	Piping & Instrumentation Diagram
PNL	Panel
PT	Instrument Potential Transformer
QSL	Qualified Suppliers List
RFI	Radio Frequency Interference
RIS	Regulatory Issues Summary
RPS	Reactor Protection System
RTD	Resistance Temperature Detector
RTS	Reactor Trip System
SWC	Normal Service Water
SWG	Switchgear
SWP	Standby Service Water
TC	Temperature Controller
TI	Temperature Indicator
TS	Temperature Switch
USAR / UFSAR	Updated Final Safety Analysis Report
V&V	Verification & Validation

I. OVERVIEW / SIGNATURES[1]

Facility: River Bend Station (RBS)

Evaluation # EN-2015-003
Rev. 0

Proposed Change / Document: EC 55881 - UHS Minimum Tech Spec Level Increase and UHS Heat Load Calculation Revisions

Description of Change:

The purpose of Evaluation Engineering Change (EC) 55881 is to allow an increase in the UHS (Ultimate Heat Sink) water storage basin inventory margin. The increase in the UHS water storage basin inventory margin is accomplished by revising calculations PM-194, PM-199, and G13.18.13.2*086. Specifically, additional UHS inventory margin is gained by revising the methodology used to determine the UHS evaporative water losses in Calculation PM-194 and by increasing the UHS Technical Specifications (TS) minimum water level from 111'-10" to 114'-5". Implementation of the proposed activity maintains compliance of the UHS with Regulatory Guide (RG) 1.27, Revision 2, and RBS USAR 9.2.5, and allows crediting the margin gained to account for leakage. The goal is to support an allowable leakage of at least 15 gpm in the SSW system while maintaining the required basin water inventory for 30 days following a design basis accident.

The Process Applicability Determination (PAD) identified that this activity involves use of an alternative evaluation methodology (UHSSIM) for establishing the design basis of the UHS. NEI 96-07, Revision 1, Section 4.2.1.3, states: "Proposed use of an alternative method is considered an adverse change that must be evaluated under 10 CFR 50.59(c)(2)(viii)." Therefore, the proposed activity adversely affects a method of evaluation that demonstrates intended design function(s) of an SSC will be accomplished as described in the UFSAR.

Summary of Evaluation: The change addressed by this evaluation is the proposed use of the Sargent & Lundy LLC Ultimate Heat Sink simulator (UHSSIM) computer program for analysis of the RBS UHS to demonstrate compliance with Regulatory Guide (RG) 1.27, Revision 2. The program is used to calculate the UHS inventory loss for the 30-day period following a design basis accident. This change is considered an alternative evaluation methodology for establishing the design basis of the UHS as described in the USAR. The UHSSIM program was originally developed by the University of Illinois, under a contract with the NRC, specifically for analysis of UHS forced draft cooling towers. The NRC approved the use of this methodology by the licensee in their evaluation of the Vogtle Units 1&2 UHS. It was also used by the NRC to evaluate and accept the RBS UHS design, as described in Supplemental Safety Evaluation Report No. 1, NUREG-0989 (SSER1). RG 1.27 states that, for evaluation of the UHS, the meteorological conditions resulting in the maximum evaporation and drift should be the worst 30-day average combination of controlling parameters based on regional climatological information. The UHSSIM program has weather search features that were used to search US National Climatic Data Center weather data recorded at the Baton Rouge Metro Airport. The UHSSIM analysis capabilities were used to identify which 30-day summer period will produce the worst case evaporation. The S&L UHSSIM program used in the UHS analysis also includes computational enhancements that were not included in the original version developed for the NRC. The proposed change to use the S&L UHSSIM program with enhanced features for analysis of the RBS UHS was evaluated to ensure it provides conservative results (i.e., the S&L UHSSIM program

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

calculates that there is less available basin water inventory at 30 days after a design basis accident as compared to the original UHSSIM program results). It was determined that this change does not result in a departure from a method of evaluation described in the USAR that was used in establishing the design bases or in the safety analyses. It is concluded, therefore, that this methodology change does not require prior NRC approval via a license amendment request.

A change to TS SR 3.7.1.1 to increase the UHS water storage basin minimum water level requires submittal of a license amendment request and NRC approval in accordance with the PAD.

Is the validity of this Evaluation dependent on any other change? ☐ **Yes** ☒ **No**

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ **Yes** ☒ **No**

Preparer: Scott Boeing/ [signature] /S&L / Engineering / 10-8-15
Name (print) / Signature / Company / Department / Date

Reviewer: Ed DeWeese/ [signature] /Entergy / Eng Design and Programs / 10-8-2015
Name (print) / Signature / Company/ Department / Date

OSRC: Todd Brumfield / [signature] / 10/12/2015
Chairman's Name (print) / Signature / Date

2015-010
OSRC Meeting #

II. 50.59 EVALUATION

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☒ Yes ☐ No

Does the proposed Change:

1. **Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR?** ☐ Yes ☐ No

   BASIS:

   N/A

2. **Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?** ☐ Yes ☐ No

   BASIS: N/A

3. **Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR?** ☐ Yes ☐ No

   BASIS: N/A

4. **Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?** ☐ Yes ☐ No

   BASIS: N/A

5. **Create a possibility for an accident of a different type than any previously evaluated in the UFSAR?** ☐ Yes ☐ No

   BASIS: N/A

6. **Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR?** ☐ Yes ☐ No

   BASIS: N/A

7. **Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered?** ☐ Yes ☐ No

   BASIS: N/A

8. **Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses?** ☐ Yes ☒ No

   BASIS:

   NEI 96-07, Rev. 1, Section 3.4, defines departure from a method of evaluation described in the UFSAR as: (i) changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same; or (ii) changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application.

   The proposed change to the method of evaluation consists of two actions as follows: 1) Replace the method currently described in the USAR for calculating 30-day UHS evaporative losses following a design-basis accident (DBA) with a method (UHSSIM program) that has been approved by the NRC for that application based on use of the program for Vogtle Units 1&2 and, 2) Use an enhanced version of the UHSSIM program that is maintained by Sargent & Lundy LLC which changes several elements of the NRC-approved version of UHSSIM, each of which is shown to produce results that are conservative or essentially the same.

   1) NRC-Approved Methodology (UHSSIM)

   Calculation PM-194 provides the RBS design basis analysis that documents the compliance of the UHS with the requirements of Regulatory Guide (RG) 1.27, Revision 2. Calculation G13.18.13.2*086 (Maximum Safeguards) considers UHS analysis scenarios beyond what is required by the RBS licensing basis for confirming safe shutdown capability following a DBA. The revisions to calculations PM-194 and G13.18.13.2*086 for this Evaluation EC incorporate new weather data and a more detailed method of evaluating UHS evaporation following a DBA. Specifically, the Ultimate Heat Sink Simulator (UHSSIM) computer program, maintained by

Sargent & Lundy LLC (S&L), is used to calculate the UHS inventory loss for the 30-day period following a DBA. The UHSSIM program provides a tool for predicting the transient temperature, mass, and dissolved solids content of power plant ultimate heat sinks that consist of a basin and cooling tower. Its algorithm considers an analysis of the cooling tower heat and mass transfer characteristics coupled with the mass and energy balance for the basin. It simulates the response of cooling towers and basins to varying weather conditions during and following a postulated accident.

The UHSSIM computer program methodology was originally developed by the University of Illinois, under a contract with the NRC, specifically for evaluating UHS forced draft cooling tower designs for compliance with Regulatory Guide (RG) 1.27. The UHSSIM program (Dunn and Sullivan Method) is docketed as ML12146A145. The NRC and industry have used this methodology to evaluate UHS designs.

In 2011, Vogtle Units 1&2 submitted a request to modify the TS (ML112450171) to increase the wet bulb temperature limit for operation of three out of four cooling tower fans. The analysis performed to support the change used the UHSSIM program maintained by Bechtel Corporation. The TS change was not submitted as a methodology change, since UHSSIM had been used previously for plant licensing. In addition, the objective of the analysis was to calculate the cooling water return temperature with the new temperature limit; basin inventory loss due to evaporation was not reanalyzed. In the 2013 NRC SER that approved the TS change (ML13231A054), the NRC stated that they performed a detailed review of the licensee's assumptions, design inputs, and methodology and compared the results to design and licensing basis requirements. This SER is considered to represent NRC approval of the UHSSIM methodology.

The NRC also used this methodology for evaluation and acceptance of the RBS UHS design, as documented in Supplemental Safety Evaluation Report No. 1, NUREG-0989 (SSER1), Section 2.4.11.2. In addition, the NRC used the methodology to evaluate the Vogtle Units 1&2 UHS to support the operating licensing review of the plant (NUREG-1137, SER Appendix F, 1985).

RBS is committed to RG 1.27, Rev. 2 (Ultimate Heat Sink for Nuclear Power Plants) which does not discuss methods of analysis for UHS evaporative cooling tower performance. Although not credited in this 50.59, Draft Regulatory Guide DG-1275 (Proposed Revision 3 for Regulatory Guide 1.27) [ML13043A624] refers to the Dunn and Sullivan method (UHSSIM) documented in ML12146A145 for use in UHS design evaluation.

The proposed methodology change for RBS is consistent with the definition of NEI 96-07, Section 3.4 (ii) provided above. NEI 96-07 additionally states that a new method of evaluation is approved by the NRC for the intended application if it is approved for the type of analysis being conducted, and applicable terms, conditions and limitations for its use are satisfied. As further described in NEI 96-07, approval of a methodology by the NRC, along with any limitations on use, is typically documented in a generic Safety Evaluation Report (SER) or a plant-specific SER and associated correspondence.

The UHSSIM program was developed for the NRC specifically for analysis of UHS forced draft cooling towers, so the proposed use by RBS is consistent with the intended application. However, because the program was developed for the NRC by the University of Illinois, a generic Safety Evaluation Report (SER) describing applicable terms, conditions, and limitations for its use has not been issued.

In the Vogtle Units 1&2 NRC SER Appendix F, 1985 and the SER for the Vogtle TS change

discussed above, there are no specific limitations identified for use of the UHSSIM methodology. Additionally, it is noted that the Vogtle TS change analysis focused on evaluation of basin water temperature, and did not evaluate basin evaporation losses because the existing licensing basis analysis of evaporation was considered conservative for the proposed change. The difference in the use of UHSSIM for the Vogtle analysis and the RBS analysis performed for this EC relates to the inputs selected to determine the highest water temperature (Vogtle) versus the highest evaporation (RBS). However, the UHSSIM methodology is the same in both cases.

Similarly, the RBS SSER1 does not describe limitations on use of the UHSSIM methodology. It does state that the worst case meteorology for the historical record of July 1948 through December 1982 was used and it was assumed that only one-half of the SCT fan cells were in operation. The analysis performed for this EC uses the UHSSIM weather search capabilities to determine the worst case meteorology for the historical record of July 2, 1948 to November 29, 2014, which is a broader date range, but includes the records used in the SSER1 analysis. The analysis for this EC also assumes that one-half the fan cells are in operation.

For the revised RBS UHS analysis, the weather search capabilities of the enhanced UHSSIM program were used to identify which 30-day summer period would produce the most evaporation. The US National Climatic Data Center (NCDC) of the National Oceanographic and Atmospheric Administration (NOAA) publishes historic hourly weather observations for the United States. RBS is approximately 20 miles north of the Baton Rouge, Louisiana Airport (KBTR), for which the NCDC has digital hourly weather records back to July 2, 1948. Raw weather observations for the Baton Rouge Metro Airport were obtained from the NCDC for the time period of July 2, 1948 to November 29, 2014. During the time period between January 1, 1965 and December 31, 1972, digital data was available every three hours rather than hourly. This data was formatted to allow UHSSIM to process it and search for the time periods that gave the largest evaporation potential, based on the RBS heat load profile and tower performance. The multiple search methods available in UHSSIM were used to assure that the selected window was the worst for evaporation. Six target windows were identified and were run in the tower simulation to compare the evaporation potential. The time period starting on June 6, 2009 was found to result in the highest evaporation from the UHS after a postulated accident. This is consistent with RG 1.27, Revision 2, which states that the meteorological conditions resulting in the maximum evaporation and drift should be the worst 30-day average combination of controlling parameters. The revision to PM-194, which is used to demonstrate RBS compliance with RG 1.27, assumed failure of the Division II diesel generator, which results in loss of one division (or one-half) of SCT fans, consistent with the current USAR analysis.

Therefore, based on previous NRC acceptance of the Dunn and Sullivan evaluation methodology contained in UHSSIM for demonstrating compliance with Regulatory Guide 1.27, Revision 2, use of this methodology to support the changes in this EC is not considered a departure from a method of evaluation described in the USAR used in establishing the design bases.

2) S&L Enhanced Version of UHSSIM

As described in the EC, the algorithms contained in the original version of UHSSIM (i.e., Dunn and Sullivan) have been enhanced in the program version used by S&L. These enhancements provide more stringent convergence criteria and diagnostics to confirm that the range of cooling tower characteristic (KaV/L) factors is large enough to assure convergence. It also allows for finer division of the tower in the computer model and includes additional physical phenomena.

Although the aforementioned UHSSIM program changes made in developing the S&L version

are considered enhancements, the program changes that were applied in the PM-194 UHS analysis were evaluated to determine if they represent a departure from a method of evaluation. NEI 96-07, Section 3.4, item (i) defines departure from a method of evaluation described in the UFSAR as: changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same.

In order to demonstrate that the S&L UHSSIM program enhancements provide conservative or essentially the same results, the following assessment was provided:

a) In the program validation performed for the S&L UHSSIM Program 03.7.870-1.0 (11/16/12), the Vogtle Units 1&2 UHS analysis contained in Appendix F of the NRC SER (NUREG-1137, April 1985) was used to recreate the analysis in UHSSIM, without use of the enhanced features, to confirm that the S&L version of the program can reproduce the results in the SER. The results of the simulation closely matched those documented in Appendix F of the SER. This shows that the S&L version of the code matches that used for the Vogtle SER analysis, and that the enhancements implemented by S&L did not negatively impact the capability of the original algorithm developed by Dunn and Sullivan.

b) A review of the methodology changes in the S&L enhanced UHSSIM as compared to the NRC UHSSIM was performed to confirm they provide results that are conservative or essentially the same.

Convergence Parameters
In Calculation PM-194, Rev. 10, performed for EC 55881, the RBS SCT model with new weather data was analyzed in UHSSIM without use of the enhanced convergence feature (i.e., the same convergence method as NRC UHSSIM). The results from this analysis were then compared to analysis results using the same model, but with the S&L UHSSIM enhanced convergence parameters and finer division of the tower model. The program inputs were the same in both cases. The comparison of calculated UHS basin inventory at 719 hours (30 days minus 1 hour) for the NRC UHSSIM and S&L enhanced program is provided below, assuming an initial water level of 114'-5" (proposed new TS minimum level). The basis for terminating the runs at 719 hours is provided in PM-194, but is not relevant to the comparison of program results. The analysis results using the S&L enhanced program are conservative.

Program	Basin Water Mass (lbm)	Basin Water Volume (gallons @ 76.5°F)
NRC UHSSIM	8.096E+06	972,532
NRC UHSSIM with S&L enhanced convergence parameters.	7.493E+06	900,196

Basin Wall Heat Transfer
The S&L enhanced UHSSIM can also determine basin wall heat transfer that results in heat addition or loss from the basin water, which is a feature that is not available in NRC UHSSIM. However, this feature was not used in the revised RBS analysis. Instead, a conservative value of wall heat transfer into the basin water was assumed as an input parameter selected by the user. Adding this heat load input to calculate forced evaporation provides more conservative results.

Leakage
Penetration Valve Leakage Control (LSV) Air Compressor leakage is described in USAR Section

9.2.5 and represents an additional cause of basin inventory loss, but cannot be modeled in the NRC UHSSIM program. The S&L UHSSIM program methodology accounts for this input, however, it is user-selected and not determined by the program algorithms. Considered independently, this input provides more conservative results since it reduces basin inventory margin.

Natural Evaporation and Tower Drift
Natural evaporation and tower drift are described in USAR Section 9.2.5 and represent additional causes of basin inventory loss, but cannot be modeled in the NRC UHSSIM program. The S&L UHSSIM program methodology accounts for these inputs, however, they are user-selected and not determined by the program algorithms. Considered independently, each of these inputs provides more conservative results since they reduce basin inventory margin.

Thermodynamic Properties for Water and Air
The revised UHS analysis also used updated thermodynamic properties for water and air based on current ASME and ASHRAE standards, since they are considered more accurate than the values used in the original NRC UHSSIM. However, water and air properties are used as an input in both NRC UHSSIM and S&L UHSSIM and can be defined by the user in both programs. Therefore, they are considered input parameters and changes to these parameters are not considered a change to a method of evaluation.

Weather Search Tool
The weather search tool used to search weather data in NRC UHSSIM uses an algorithm that is time-efficient, but it prevents the search windows from overlapping and limits the data range. The S&L enhanced UHSSIM uses a weather search algorithm that results in a more rigorous search of weather data to find the worst 30-day window. The enhanced method improves the ability to select conservative weather input data, but the results are considered essentially the same.

Diagnostic Features
The diagnostic features in the S&L enhanced program provide the user with additional information to support analysis, but do not affect the algorithms or alter input parameters used in the computations. Therefore, the presence of the diagnostic features provides results that are essentially the same.

Based on the foregoing, the comparison of the individual program features utilized in the S&L enhanced UHSSIM vs. NRC UHSSIM shows that the enhanced program provides results that are "conservative or essentially the same". Overall, the use of the S&L UHSSIM program with enhanced features in this specific application results in a higher evaporation rate and less basin inventory (i.e., less margin) at 30 days, assuming an initial water level of 114'-5", as documented in PM-194:

Program	Basin Water Mass (lbm)	Basin Water Volume (gallons @ 76.5°F)
Bounding RBS UHSSIM Model – S&L UHSSIM with enhanced convergence, and updated thermodynamics	6.077E+06	730,000

Therefore, use of this program for the UHS analysis is not a departure from a method of evaluation, per the NEI 96-07, Section 3.4 guidance.

NEI 96-07, Section 4.3.8.2, contains guidance for evaluating use of a new method of evaluation. This guidance is repeated below with responses in italics pertaining to the proposed change for RBS.

"Considerations for Determining if New Methods May be Considered "Approved by the NRC for the Intended Application"

The following questions highlight important considerations for determining that a particular application of a different method is technically appropriate for the intended application, within the bounds of what has been found acceptable by NRC, and does not require prior NRC approval."

- "Is the application of the methodology consistent with the facility's licensing basis (e.g., NUREG-0800 or other plant-specific commitments)?"

  Response: Yes. The RBS UHS design basis complies with NRC RG 1.27, Rev. 2, Ultimate Heat Sink for Nuclear Power Plants. This RG requires that the meteorological conditions resulting in maximum evaporation and drift losses should be the worst 30-day average combination of controlling parameters. The UHSSIM methodology developed by Dunn and Sullivan (ML12146A145) specifically addresses this RG requirement. The S&L enhanced version of UHSSIM also directly addresses this requirement. Therefore, use of the program is consistent with the RBS licensing basis.

- "Will the methodology supersede a methodology addressed by other regulations such as 10 CFR 50.46, 10 CFR 50.55a or the plant technical specifications (Core Operating Limits Report or Pressure/ Temperature Limits Report)?"

  Response: No. The new methodology will supersede the existing USAR-described methodology used to calculate SCT basin evaporation following a DBA, however, the existing methodology is a plant-specific method that is used to show compliance with RG 1.27 and is not addressed by other regulations or the TS.

- "Is the methodology consistent with relevant industry standards?"

  Response: Yes. The Dunn and Sullivan methodology was developed uniquely for UHS analysis to demonstrate compliance with NRC RG 1.27. Although it is not based on specific industry standards, UHSSIM does use the Merkel cooling tower characteristic parameter (KaV/L) which is widely used in evaluating cooling tower performance. Dunn and Sullivan also considered results from published research papers in development of the program. The enhancements in the S&L version of UHSSIM were also not based on a specific industry standard, but inputs from recognized published sources such as ASHRAE and ASME were utilized.

- "If a computer code is involved, has the code been installed in accordance with applicable software quality assurance requirements?"

  Response: Yes. The S&L UHSSIM Program 03.7.870-1.0 has been validated and approved for use in accordance with the S&L Quality Assurance (QA) program. The RBS UHS analysis was performed on a PC running the Windows 7 operating system.

- "Has the plant specific model been adequately qualified through benchmark comparisons against test data, plant data or approved engineering analyses?"

  Response: Yes. The RBS plant-specific SCT model is documented in approved engineering

Calculations PM-194 and PM-199. PM-194 is revised to use the methodology in the UHSSIM program to analyze the existing SCT model in order to calculate basin evaporative water losses. In addition, a benchmark comparison of the S&L UHSSIM program was performed against the Vogtle UHS design.

- "Is the application consistent with the capabilities and limitations of the computer code?"

  Response: Yes. The Dunn and Sullivan method formulates the cooling tower analysis based on counterflow direct contact water cooling by air, which is representative of a forced draft cooling tower. The test case analyzed in the Dunn and Sullivan paper (ML12146A145) is a circular, forced draft cooling tower with characteristics that are very similar to the RBS SCT. This includes such parameters as number of cells, number of fans per cell, basin volume, cooling water flowrate, maximum return temperature, and design wet bulb temperature. The test case also assumes that one train (two cells) is in operation, as in the RBS analysis. The application for RBS is clearly within the capabilities and limitations of the program. The S&L version of UHSSIM adds enhancements, but does not reduce the capabilities of the original program.

- "Has industry experience with the computer code been appropriately considered?"

  Response: Yes. See previous discussion regarding NRC and Vogtle use of UHSSIM to evaluate UHS performance for RBS and Vogtle.

- "Is the facility for which the methodology has been approved designed and operated in the same manner as the facility to which the methodology is to be applied?"

  Response: Yes. The Vogtle 1&2 UHS cooling towers are also mechanical forced draft towers. The Vogtle towers also have circular basins. However, for each unit, there are two separate, four cell, four fan towers, one for each train. The individual tower sizes are smaller than the single RBS tower, however, the maximum cooling water return temperature and design wet bulb temperatures are nearly the same. Both the Vogtle and RBS UHSs are designed to comply with RG 1.27, Rev. 2.

- "Is the relevant equipment the same? Does the equipment have the same pedigree (e.g., Class 1E, Seismic Category I, etc.)? Are the relevant failure modes and effects analyses the same? If the plant is designed and operated in a similar, but not identical, manner, the following types of considerations should be addressed to assess the applicability of the methodology:
  - How could those differences affect the methodology?
  - Are additional sensitivity studies required?
  - Should additional single failure scenarios be considered?
  - Are analyses of limiting scenarios, effects of equipment failures, etc., applicable for the specific plant design?
  - Can analyses be made while maintaining compliance with both the intent and literal definition of the methodology?"

  Response: Yes. The relevant equipment is the same. The cooling towers at both plants are designed as safety-related and operated to perform the same design function. Relevant failure modes applicable to the analysis are the same. The analyses for both plants assume a single failure, which is the loss of one train of tower fans at RBS, or loss of one of two towers at Vogtle. The S&L UHSSIM program validation performed in 2012 replicated the Vogtle UHS analysis described in Appendix F of the Vogtle SER (NUREG-1137, April 1985) to confirm that the S&L version of the program could reproduce the results in the SER. No additional sensitivity studies are required.

- "Differences in the plant configurations and licensing bases could invalidate the application of a particular methodology. For example, the licensing basis of older vintage plants may not include an analysis of the feedwater line break event that is required in later vintage plants. Some plants may be required to postulate a loss of off-site power or a maximum break size for certain events; others may have obtained exemptions to these requirements from the NRC. Some plants may have pressurizer power-operated relief valves that are qualified for water relief; other plants do not. Plant specific

failure modes and effects analyses may reveal new potential single failure scenarios that cannot be adequately assessed with the original methodology. The existence of these differences does not preclude application of a new methodology to a facility; however, differences must be identified, understood and the basis documented for concluding that the differences are not relevant to determining that the new application is technically appropriate."

Response: Not applicable. The Dunn and Sullivan methodology was not developed for a plant-specific application. It was developed to provide a standardized method for analyzing the performance of UHS forced draft cooling towers. Also, as stated in the Dunn and Sullivan paper (ML12146A145), their objective was to develop a realistically conservative analysis method which is flexible enough to handle any proposed UHS cooling tower design and is easy to apply to many different sites. Likewise, the enhanced features in the S&L UHSSIM program were not developed for a plant-specific application. There are no RBS plant-specific UHS design features or failure modes that would invalidate use of the new methodology.

Based on the foregoing, it is concluded that the proposed change to use the S&L UHSSIM program, with enhance features, for analysis of the RBS UHS does not result in a departure from a method of evaluation described in the USAR that was used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

I. OVERVIEW / SIGNATURES[1]

Facility: River Bend Station **Evaluation # / Rev. #:** 0

Proposed Change / Document: EC58840 EN-2015-004

Description of Change:

The change being evaluated is to increase the Main Steam Isolation Valve - Closure Reactor Protection System (RPS) Instrumentation Function (TRM Table 3.3.1.1-1, Function 6) response time from ≤ 0.09 secs to ≤ 0.150 secs being implemented by EC58840.

Summary of Evaluation:

Evaluation of the activity within the scope of this EC, including the revised analyses associated with the change, determined that for the eight criteria of 10CFR50.59(c)(2):

(i) There will not be more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR because the Reactor Protection System (RPS) Instrumentation Main Steam Isolation Valve - Closure Function logic affected by this change is passive during normal plant operation and would not change the possible causes for the MSIV Closure event and the change would not affect those evaluated contributors to the frequency of the MSIV Closure event.

(ii) There will not be more than a minimal increase in likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the USAR because there is no change in the circuit's vulnerability to malfunctions or change in vulnerability to malfunctions of other SSCs either directly or indirectly because there is no new hardware or circuitry being added and no physical changes are being made to existing hardware or circuitry.

(iii) There will not be more than a minimal increase in the consequences of an accident previously evaluated in the USAR. GEH analysis does not reflect any increase of radiological consequences as a result of reanalysis of the event to evaluate the impact of the change to the RPS MSIV Closure logic. Additionally, the conservative assumptions utilized in the analysis documented in USAR Section 15.2.4.5 remain unchanged and bounding for the event.

(iv) There will not be more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR. The change to the RPS MSIV Closure logic response time does not affect the analysis of the radiological consequences for the event or the postulated contributing malfunctions.

(v) There will not be a possibility for an accident of a different type than any previously evaluated in the USAR because there is no change in the circuit's vulnerability to malfunctions or change in vulnerability to malfunctions of other SSCs either directly or indirectly that would cause an accident because there is no new hardware or circuitry being added and no physical change to existing hardware or circuitry. Since the circuit is passive during normal plant operation, the change in response time of the circuit can have no impact on SSCs important to safety during normal plant operation. The circuit is only active during the MSIV Closure event.

(vi) A possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in USAR will not be created because there is no change in the circuit's vulnerability to malfunctions or change in vulnerability to malfunctions of other SSCs either directly or indirectly that would cause the plant or an SSC to respond in a different manner than currently evaluated in the USAR because there is no new hardware or circuitry being added and no physical change to existing hardware or circuitry. Since the circuit is passive during normal plant operation, the change in response time of the circuit can have no impact on SSCs important to safety during normal plant operation. The circuit is only active during the MSIV Closure event and the change to the circuit has been evaluated to show that the response of the plant remains bounded by conditions of other events.

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

(vii) A design basis limit for a fission product barrier as described in the USAR will not be exceeded or altered because an updated analysis for the event to evaluate the impact of the change on the USAR evaluated transient shows that reactor pressure stays within design limits and remains bounded by other pressurization events

(viii) There is no departure from a method of evaluation described in the USAR that is used to establish the design basis or used in the safety analyses because the activity does not involve replacing any USAR described evaluation methodologies.

Is the validity of this Evaluation dependent on any other change? ☐ **Yes** ☒ **No**

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ **Yes** ☒ **No**

Preparer:	WH Chenault	/ See AS	/ Enercon	/ Eng & Safety Analysis	/ 9/23/15
	Name (print)	/ Signature	/ Company	/ Department	/ Date
Reviewer:	Bivins Calhoun	/ See AS	/ Enercon	/Mechanical Engineering	/ 9/28/15
	Name (print)	/ Signature	/ Company	/ Department	/ Date
OSRC:	Todd Brumfield	/ Todd Brumfield			/ 10/12/2015
	Chairman's Name (print)	/ Signature			/ Date

2015-010
OSRC Meeting #

II. 50.59 EVALUATION

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:

The change being evaluated is to increase the Reactor Protection System (RPS) Instrumentation Main Steam Isolation Valve - Closure Function (TRM Table 3.3.1.1-1, Function 6) response time from ≤ 0.09 secs to ≤ 0.150 secs. The RPS MSIV Closure response time is associated with the response of the plant to an event (MSIVs closing when the reactor mode switch is in RUN) evaluated in the accident analysis of the USAR (Chapter 15 analysis). However, the response time is associated with the circuit dropping out after the event (MSIVs closing) has occurred. The circuit is passive prior to the event occurring.

USAR Section 15.2.4.1.1, Identification of Causes, describes possible causes for the MSIV Closure event; these include various steam line and nuclear system malfunctions, or operator actions. Examples are low steam line pressure, high steam line flow, low water level, or manual action. USAR Section 15.2.4.1.2, Frequency Classification, describes what contribute to the frequency of the MSIV closure event, they are: manual action (purposely or inadvertent); spurious signals such as low pressure, low reactor water level, low condenser vacuum and finally, equipment malfunctions such as faulty valves or operating mechanisms. The RPS MSIV Closure logic circuits affected by this change would not change the possible causes for the MSIV Closure event and the change would not affect those evaluated contributors to the frequency of the MSIV Closure event.

Additionally, the RPS MSIV Closure logic circuits affected by this change do not initiate any accidents currently evaluated within the USAR. The change made by this EC does not create any new failure modes for the affected RPS MSIV Closure circuits such that they would be able to initiate a USAR described accident. The affected RPS MSIV Closure circuits will remain fully capable of performing their intended design function subsequent to the change. Therefore, as discussed above, the change does not change any accident initiators previously evaluated in the USAR or introduce new accident initiators not previously evaluated in the USAR. Since the affected logic circuits associated with this activity are unable to initiate any USAR described accidents, the activity does not result in an increase to the frequency of occurrence of any accidents previously evaluated in the USAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

The change being evaluated is to increase the RPS Instrumentation Main Steam Isolation Valve - Closure Function (TRM Table 3.3.1.1-1, Function 6) response time from ≤ 0.09 secs to ≤ 0.150 secs. The function of this RPS MSIV Closure circuit is to mitigate an event evaluated in USAR Section 15.2.4, MSIV Closure and more specifically described in USAR Section 15.2.4.1.2.1, Closure of All Main Steam Isolation Valves. To mitigate reactor pressure increase for this event, position switches on the MSIVs and the protection system provide a reactor scram if the MSIVs in 3 or more main steam lines are less than an instrument setpoint normally set within the range between the nominal setpoint of 92% open and the allowable setpoint of 88% open (the analytical limit is 85% open, USAR Section 15.2.4.3.2). Relief valves also operate to limit system pressure. All these aspects are designed to single failure criterion and additional single failures would not alter the results of this analysis (USAR Section 15.2.4.2.3, The Effect of Single Failures and Operator Errors).

USAR Section 15.2.4.4.1, Barrier Performance Closure of All Main Steam Line Isolation Valves, currently states: "*The nuclear system relief valves begin to open at approximately 3.6 sec after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak* pressure *at the vessel bottom reaches 1,259 psig, clearly below the pressure limits of the RCPB. Peak pressure in the main steam line is 1,224 psig.*"

For the overpressurization protection analysis of USAR Section 5.2.2.2.3.1, Safety/Relief Valve Capacity, the Average Power Range Monitor Fixed Neutron Flux-High Function, along with the SRVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is assumed to fail and is not credited in the overpressurization analysis (Technical Specifications Bases 3.3.1.1). Therefore, since the MSIV closure circuit is not credited for this overpressurization analysis the change does not impact results of the existing analysis and does not affect its likelihood.

The proposed change does not affect any hardware in the affected circuits except to increase the response time from ≤ 0.09 secs to ≤ 0.150 secs. The effect of increasing the response time is an increase in time in scram initiation (maximum increase of 0.06 seconds over the current limit) following the event which results in an increase of peak reactor pressure following the event. General Electric Hitachi (GEH) updated analysis of the change in RPS MSIV Closure response time (001N8591-R1, issued as revision 1 of RBS-SA-14-00002) determined that the SRVs continue to open at approximately 3.6 seconds and that the Maximum Vessel Pressure would be 1262 psig and Maximum Steamline pressure would be 1228 psig. The GEH analysis concluded that "*The MSIVD event with a position switch setpoint of 85% and a scram delay of 0.15 seconds remains non-limiting compared to other pressurization events.*" The reactor pressure remains under the design limit of 1375 psig during this event.

Because there is no new hardware or circuitry being added and no physical change to existing hardware or circuitry, there is no change in the circuit's vulnerability to malfunctions or change in vulnerability to malfunctions of other SSCs either directly or indirectly. Therefore, it is concluded that this activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of equipment important to safety.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

The change being evaluated is to increase the RPS Instrumentation Main Steam Isolation Valve - Closure Function (TRM Table 3.3.1.1-1, Function 6) response time from ≤ 0.09 secs to ≤ 0.150 secs. The event with radiological consequences previously evaluated in the USAR associated with this change is evaluated in USAR Section 15.2.4, MSIV Closure and more specifically described in USAR Section 15.2.4.1.2.1, Closure of All Main Steam Isolation Valves. To mitigate reactor pressure increase for this event, position switches on the MSIVs and the protection system provide a reactor scram if the MSIVs in 3 or more main steam lines are less than an instrument setpoint normally set within the range between the nominal setpoint of 92% open and the allowable setpoint of 88% open (the analytical limit is 85% open, USAR Section 15.2.4.3.2). Relief valves also operate to limit system pressure.

USAR Section 15.2.4.5, Radiological Consequences, states: "*At the time of MSIV closure the containment is being purged at a rate of 7,000 cfm. Upon receipt of a high-high radiation signal from the containment purge ventilation exhaust monitor, the containment purge fans are shut down and the isolation dampers are closed. No significant amount of radioactivity is released during this time period. A conservative estimate of the maximum radioactivity concentrations which could result from MSIV closure is presented in Table 15.2-7. The analysis shows that concentrations are well within 10CFR20, Appendix B, Table II limits.*"

The proposed change, increase in the RPS MSIV Closure response time, has no impact on the high-high radiation signal used to initiate shutdown of the containment purge fans and closure of the isolation dampers because they are independent signals, therefore, there would be no change to the existing analysis regarding release of radioactivity for this event. Additionally, the conservative assumptions utilized in the analysis documented in USAR Section 15.2.4.5 remain unchanged and bounding for the event.

As concluded in EC 58840, based on the results of the GEH analysis (001N8591-R1, issued as revision 1 of RBS-SA-14-00002) there are no impacts to parameters associated with the MSIV Closure event except: Maximum Neutron Flux (% NBR); Maximum Dome Pressure (psig); Maximum Vessel Pressure (psig); Maximum Steam Line Pressure (psig); Maximum Core Average Surface Heat Flux (% Initial); and, Corrected Differential Critical Power Ratio for GNF2 Fuel. No change to radiological consequences was identified in the GEH analysis. Additionally, the GEH analysis concluded that "*The MSIVD event with a position switch setpoint of 85% and a scram delay of 0.15 seconds remains non-limiting compared to other pressurization events.*"

USAR Section 15.2.4.5.1, Fission Product Release from Fuel, currently states: "*Since each of those transients identified previously which cause SRV actuation results in various vessel depressurization and steam blow- down rates, the transient evaluated in this section is that one which maximizes the radiological consequences for all transients of this nature. This transient is the closure of all MSIVs.*" The changes to the RPS MSIV Closure response time does not change the radiological consequences of the closure of all MSIVs event. Therefore, the radiological consequences for the closure of all MSIVs event remains bounding for the events evaluated in USAR Sections 15.1.2, 15.1.3, 15.1.4, 15.2.1, 15.2.2, 15.2.3, 15.2.5, 15.2.6, 15.2.9, 15.3.1, 15.3.2, 15.3.3 and 15.3.4.

Therefore, there will not be more than a minimal increase in the consequences of an accident previously evaluated in the USAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:

USAR Section 15.2.4.1.1, Identification of Causes, describes possible causes for the MSIV Closure event; these include various steam line and nuclear system malfunctions, or operator actions. Examples are low steam line pressure, high steam line flow, low water level, or manual action. USAR Section 15.2.4.1.2, Frequency Classification, describes what contribute to the frequency of the MSIV closure event, they are: manual action (purposely or inadvertent); spurious signals such as low pressure, low reactor water level, low condenser vacuum and finally, equipment malfunctions such as faulty valves or operating mechanisms. The malfunctions described in USAR Sections 15.2.4.1.1 and 15.2.4.1.2 which are postulated to cause the MSIV Closure event have been evaluated for their radiological consequences. As discussed in in the basis for Question 3 above, the change to the RPS MSIV Closure logic response time does not affect the analysis of the radiological consequences for the event or the postulated contributing malfunctions because these signals are independent. Therefore, this activity does not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR

5. Create a possibility for an accident of a different type than any previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:

The change being evaluated is to increase the Reactor Protection System (RPS) Instrumentation Main Steam Isolation Valve - Closure Function (TRM Table 3.3.1.1-1, Function 6) response time from ≤ 0.09 secs to ≤ 0.150 secs. The RPS MSIV Closure response time is associated with the response of the plant to an event evaluated in the accident analysis of the USAR (Chapter 15 analysis). However, the response time is associated with the circuit dropping out after the event (MSIVs closing) has occurred. The circuit is passive prior to the event occurring. The proposed change does not affect any hardware in the affected circuits except to increase the response time from ≤ 0.09 secs to ≤ 0.150 secs. The effect of increasing the response time is an increase in time in scram initiation following the event which results in an increase of peak reactor pressure following the event. GEH updated analysis of the change in RPS MSIV Closure response time (001N8591-R1, issued as revision 1 of RBS-SA-14-00002). The GEH analysis concluded that "*The MSIVD event with a position switch setpoint of 85% and a scram delay of 0.15 seconds remains non-limiting compared to other pressurization events.*"

Because there is no new hardware or circuitry being added and no physical change to existing hardware or circuitry, there is no change in the circuit's vulnerability to malfunctions or change in vulnerability to malfunctions of other SSCs either directly or indirectly that would cause an accident. Since the circuit is passive during normal plant operation, the change in response time of the circuit can have no impact on SSCs important to safety during normal plant operation. The circuit is only active during the MSIV Closure event. Therefore, this activity does not create a possibility for an accident of a different type than any previously evaluated in the USAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

The change being evaluated is to increase the Reactor Protection System (RPS) Instrumentation Main Steam Isolation Valve - Closure Function (TRM Table 3.3.1.1-1, Function 6) response time from ≤ 0.09 secs to ≤ 0.150 secs. The RPS MSIV Closure response time is associated with the response of the plant to an event evaluated in the accident analysis of the USAR (Chapter 15 analysis). However, the response time is associated with the circuit dropping out after the event (MSIVs closing) has occurred. The circuit is passive prior to the event occurring. The proposed change does not affect any hardware in the affected circuits except to increase the response time from ≤ 0.09 secs to ≤ 0.150 secs. The effect of increasing the response time is an increase in time in scram initiation following the event which results in an increase of peak reactor pressure following the event.

As concluded in EC 58840, based on the results of the GEH analysis (001N8591-R1, issued as revision 1 of RBS-SA-14-00002) there are no impacts to parameters associated with the MSIV Closure event except: Maximum Neutron Flux (% NBR); Maximum Dome Pressure (psig); Maximum Vessel Pressure (psig); Maximum Steam Line Pressure (psig); Maximum Core Average Surface Heat Flux (% Initial); and, Corrected Differential Critical Power Ratio for GNF2 Fuel. No adverse effects were identified in the GEH analysis. Additionally, the GEH analysis concluded that "*The MSIVD event with a position switch setpoint of 85% and a scram delay of 0.15 seconds remains non-limiting compared to other pressurization events.*"

Because there is no new hardware or circuitry being added and no physical change to existing hardware or circuitry, there is no change in the circuit's vulnerability to malfunctions or change in vulnerability to malfunctions of other SSCs either directly or indirectly that would cause the plant or an SSC to respond in a different manner than currently evaluated in the USAR. Since the circuit is passive during normal plant operation, the change in response time of the circuit can have no impact on SSCs important to safety during normal plant operation. The circuit is only active during the MSIV Closure event and the change to the circuit has been evaluated to show that the response of the plant remains bounded by conditions of other events. Therefore, this activity does not create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the USAR.

7. Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The change being evaluated is to increase the RPS Instrumentation Main Steam Isolation Valve - Closure Function (TRM Table 3.3.1.1-1, Function 6) response time from ≤ 0.09 secs to ≤ 0.150 secs. The function of this RPS MSIV Closure circuit is to mitigate an event evaluated in USAR Section 15.2.4, MSIV Closure and more specifically described in USAR Section 15.2.4.1.2.1, Closure of All Main Steam Isolation Valves. This event challenges the reactor coolant boundary because of the resultant pressure transient. To mitigate reactor pressure increase for this event, position switches on the MSIVs and the protection system provide a reactor scram if the MSIVs in 3 or more main steam lines are less than an instrument setpoint normally set within the range between the nominal setpoint of 92% open and the allowable setpoint of 88% open (the analytical limit is 85% open, USAR Section 15.2.4.3.2). Relief valves also operate to limit system pressure.

USAR Section 15.2.4.4.1, Barrier Performance Closure of All Main Steam Line Isolation Valves, currently states: "*The nuclear system relief valves begin to open at approximately 3.6 sec after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom reaches 1,259 psig, clearly below the pressure limits of the RCPB. Peak pressure in the main steam line is 1,224 psig.*"

The overpressurization protection analysis of USAR Section 5.2.2.2.3.1, Safety/Relief Valve Capacity, the Average Power Range Monitor Fixed Neutron Flux-High Function, along with the SRVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is assumed to fail and is not credited in the overpressurization analysis.

The proposed change does not affect any hardware in the affected circuits except to increase the response time from ≤ 0.09 secs to ≤ 0.150 secs. The effect of increasing the response time is an increase in time in scram initiation (maximum increase of 0.06 seconds over the current limit) following the event which results in an increase of peak reactor pressure following the event. GEH updated analysis of the change in RPS MSIV Closure response time (001N8591-R1, issued as revision 1 of RBS-SA-14-00002) determined that the SRVs continue to open at approximately 3.6 seconds and that the Maximum Vessel Pressure would be 1262 psig and Maximum Steamline pressure would be 1228 psig. The GEH analysis concluded that "*The MSIVD event with a position switch setpoint of 85% and a scram delay of 0.15 seconds remains non-limiting compared to other pressurization events.*" The reactor pressure remains under the design limit of 1375 psig during this event.

Therefore, based on updated analysis for the event to evaluate the impact of the change on the USAR evaluated transient which shows that reactor pressure stays within design limits and remains bounded by other pressurization events, a design basis limit for a fission product barrier as described in the USAR will not be exceeded or altered

8. Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS: As stated in the PAD for EC58840, the method of evaluation used to demonstrate that the intended design function of the MSIV position scram will be accomplished is based on the computer model stated in section 15.1.2.3.1 of the USAR. This model referenced is the NRC staff approved GEH ODYN software as described in NEDO-24154. The analysis contained in EC58840 also utilizes the ODYN software. The activity does not involve replacing any USAR described evaluation methodologies. Even though the activities within the scope of this change will impact the plant response following the subject MSIV closure event, the revised analyses, considering both direct and indirect effects, indicates that the changes will be minimal and that the SSCs will continued to perform their design basis function. The revised analysis does not change any element of the analysis methodology that would yield results that are non-conservative or not essentially the same as the results of the original analysis. The revised analysis does not use a new or different method of evaluation that is not approved by the NRC for the intended application. Therefore, the activity does not result in a departure from a method of evaluation described in the USAR that is used to establish the design basis or used in the safety analyses.

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

I. OVERVIEW / SIGNATURES[1]

Facility: River Bend Station **Evaluation #** 6-04/EN-2016-01 **/ Rev. #:** 0

Proposed Change / Document: EC 59951 MSO Alternate Feedwater Trip

Description of Change:

Calculation G13.18.12.2-139, Rev. 0 (Estimated Time to Overfill the RPV Due to Continued Feedwater Operation during a Fire in the Main Control Room) concludes that under normal reactor pressure conditions, the time for the Reactor Pressure Vessel (RPV) to fill is 39 seconds following a scram. A fire in the Main Control Room (MCR) can disable the trips to the three reactor feedwater pumps (FWS-P1A, FWS-P1B, and FWS-P1C). The 39 seconds to fill is insufficient time for manual action to shut down the feedwater pumps. If the RPV water level exceeds level 8 the Reactor Core Isolation Cooling (RCIC) system may become inoperable due to automatic closure of the RCIC steam supply valve and/or flooding of the RCIC steam supply piping.

EC 59951 implements a design to provide an alternate method of ensuring that the Reactor Feedwater (FWS) Pumps are tripped on increasing Reactor Pressure Vessel (RPV) level during a fire in the Main Control Room that prevents the plant operators from securing the FWS in the normal manner. The FWS pump trip function is provided by a new, independent, 2-out-of-2 logic trip circuit for the FWS pump motor circuit breakers. The input for the new circuit is provided by new level transmitters, located in the Reactor Building, providing level signals to electronic switches located in a new panel installed in the Normal Switchgear Building. The output of the trip circuit is utilized to energize new secondary trip coils added to the FWS pump motor circuit breakers, in 13.8 kV switchgear that is also located in the Normal Switchgear Building.

The RPV water level that triggers an "MSO (Multiple Spurious Operations) Level 8" trip will be purposefully elevated above existing Level 8 Trips to assure the MSO level 8 trip is sequenced as the trip-of-last-occurrence.

EC 59951 installs new control panel, C33-PNL100 on the 98 foot elevation inside the Normal Switchgear Building (NSB). Power to C33-PNL100 is fed from panels BYS-

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

PNL03A and BYS-PNL03B located in the Turbine Building (TB). Reactor Pressure Vessel (RPV) level signals are fed from transmitters C33-LTN006A and C33-LTN006B located in Reactor Building panels H22-P004 and H22-P027, respectively. Trip cards, C33-ESN606A and C33-ESN606B, inside control panel C33-PNL100 compare the RPV level signals to internally calibrated setpoints and energize relays C33A-K17A and C33A-K17B, respectively, when monitored RPV level signals exceed trip card setpoints for MSO level 8.

When relays C33A-K17A and C33A-K17B are energized concurrently and when feedwater pump circuit breaker NPS-SWG1A-ACB12 is closed, then 125VDC power sourced from BYS-PNL03A is fed to the secondary trip coil of breaker NPS-SWG1A-ACB12 to TRIP the breaker and de-energize the FWS-P1A motor. Similarly, when relays C33A-K17A and C33A-K17B are energized concurrently and when feedwater pump circuit breakers NPS-SWG1B-ACB28 or NPS-SWG1B-ACB29 are closed, 125VDC power sourced from BYS-PNL03B is fed to the secondary trip coils of breakers NPS-SWG1B-ACB28 and/or NPS-SWG1B-ACB29 (depending on breaker status) to TRIP these breakers and de-energize FWS-P1B and/or FWS-P1C motors.

During Mode 3 certain operations require level 8 to be exceeded in the Reactor Vessel so a bypass switch will be included for the feedwater pump MSO level 8 trips to allow operators the ability to prevent operation of the new trip circuits at RPV level above level 8. Additionally, enclosure 18 in Procedure EOP-0005, Emergency Operating and Severe Accident Procedures Enclosures, will be revised to add the Bypass switch function.

Electrical and civil/structural systems impacted by EC 59951 are non-safety related. However, cables newly installed by EC 59951 are credited for safe shutdown during a Main Control Room fire and the site specific Design Criterion 240.201A, Post-Fire Safe Shutdown Analysis is impacted by EC 59951. Based on changes to the site safe shutdown strategy, EC 59951 is classified as quality related.

Related ECs

Equivalent Engineering Change EQC 60495 evaluated and approved the use of refurbished breakers provisioned with new secondary trip coils for installation in cubicles NPS-SWG1A-ACB12, NPS-SWG1B-ACB28, and NPS-SWG1B-ACB29. Breakers with dual trip coils permit one coil (primary) to be connected to the existing trip logic that

originates from the MCR. The new secondary coil enables tripping of these breakers from power sources that are independent of the existing control circuits and are not impacted by control room fires.

EC 62305 installed transmitters C33-LTN006A and C33-LTN006B inside containment. Transmitter circuits were routed from the instrument cabinets and terminated in existing penetration cabinet RCP*TCR02A (located inside containment) which is connected to RCP*TCA02 (located outside containment). New cable was terminated in RCP*TCA02 and routed out of the penetration cabinet where it was coiled for use by EC 59951.

Background

The NRC evaluated the following in their 2013 Triennial Inspection of River Bend Station, as it is discussed further in Enclosure 2 to "Response to River Bend Station Disputed Violation and Errata to – NRC Triennial Fire Protection Inspection Report 05000458/2013007."

Failure to Properly Calculate the Time Available for Operator Actions

Introduction

The team identified a Green non-cited violation of License Condition 2.C.(10) for the failure to implement and maintain in effect all provisions of the approved fire protection program. Specifically, the licensee failed to properly calculate the amount of time available for operators to perform time critical actions for all control room fire scenario descriptions.

Following the 2010 triennial fire protection inspection, the licensee reviewed the design basis for determining the amount of time available for operators to perform select time critical actions in the alternative shutdown procedure. The licensee determined that several different calculations formed the design basis for determining the amount of time available.

These calculations were performed in accordance with the safe shutdown analysis and determined the amount of time available for operators to perform specific actions under various alternative shutdown scenarios. The safe shutdown analysis incorporated NRC staff guidance related to control room fire scenarios. Specifically, the safe shutdown analysis made the following two assumptions: (1) offsite power was lost as well as the automatic starting of the emergency diesel generators and (2) the automatic function of valves and pumps whose control circuits could be affected by a control room fire was lost. The safe shutdown analysis also noted that the only operator action in the control room prior to evacuation given credit was the reactor trip. Finally, the safe shutdown analysis noted that the safe shutdown capability should not be adversely affected by any one spurious action or signal resulting from a fire.

The NRC team reviewed the assumptions, methods, and results of these calculations. The NRC team identified one alternative shutdown scenario where RBS failed to properly calculate the amount of time available for operators to perform time critical actions. This scenario involved the amount of time available to terminate feedwater injection prior to overfilling the reactor vessel.

For this scenario, the NRC team noted that overfilling the reactor vessel could disable the reactor core isolation cooling system and damage the steam lines. The reactor core isolation cooling system was relied upon in this scenario to restore and maintain reactor vessel level and control pressure. Overfilling the reactor vessel could also damage the safety relief valves since they were not analyzed to pass high pressure water.

This issue was first identified during the 2010 triennial fire protection inspection as non-cited violation 05000458/2010006-03. In response to this violation, RBS revised the alternative shutdown procedure to direct the auxiliary control room operators to immediately terminate feedwater injection by closing the condensate demineralizer service inlet and outlet isolation valves. The nominal stroke time for these motor- operated valves is 48 seconds.

In addition, RBS performed Calculation G13.18.12.2-139, Rev. 0, "Estimated Time to Overfill the RPV Due to Continued Feedwater Operation During a Fire in the Main Control Room," to determine the amount of time to overfill the reactor vessel and flood the A main steam line up to the RCIC steam supply line. Based on this calculation, operators would have less than 45 seconds available to terminate feedwater injection if all three of the normally running feedwater pumps continued to inject. Based on this calculation, RBS concluded that "the overfill condition happens so quickly that manual action outside of the control room to mitigate the concern has a low probability of success." Further, RBS concluded that "directing the Auxiliary Control Room to close the condensate demineralizer filter valves and the actual closing of the valves would require more than one minute."

RBS then generated a corrective action item to perform an evaluation that could be used as a basis for a deviation request to justify the manual actions to be taken in the auxiliary control room of closing the condensate demineralizer filter valves for preventing reactor vessel overfill during a control room fire scenario with the continued injection of feedwater.

RBS subsequently revised Calculation G13.18.12.2-139 to examine scenarios where one or two feedwater pumps continued to inject and the remaining feedwater pumps stopped. The revised calculation concluded that operators would have approximately 2 minutes available to terminate feedwater injection if only one of the normally running feedwater pumps continued to inject. RBS performed a timed walk-down of these steps and concluded that operators could perform the required actions within a range of 1 minute 53 seconds to 2 minutes 15 seconds. RBS concluded that no deviation request was necessary since operators could reasonably perform the actions within the required time and the actions could not be undone by a control room fire. Further, RBS noted that this control room fire scenario involved

multiple spurious actuations.

The NRC team reviewed RBS's evaluation and concluded that RBS incorrectly implemented the guidance for control room fires contained in the safe shutdown analysis. Specifically, the NRC team noted that the continued injection of the feedwater pumps and the loss of automatic actuation/trip signals were part of the design assumptions for a control room fire and were not considered to be spurious actuations. The NRC team concluded that the bounding control room fire scenario involved the continued injection of all three normally running feedwater pumps and operators had less than 45 seconds to terminate feedwater injection.

Based on the timed walk-down of the alternative shutdown procedure, the NRC team determined that the auxiliary control room operators would complete the immediate action of terminating feedwater injection in approximately 2 minutes 5 seconds.

RBS entered this issue into their corrective action program for further evaluation and implemented enhanced operator rounds as a compensatory measure for this issue.

The failure to properly calculate the amount of time available for operators to perform time critical actions for all control room fire scenarios was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

License Condition 2.C.(10) requires that RBS comply with the requirements of their fire protection program and that RBS shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment 22 and as approved in the Safety Evaluation Report dated May 1984 and Supplement 3 dated August 1985.

Introduction

This 50.59 Evaluation examines only the new MSO level 8 Feedwater pump trip. Therefore, the only directly affected system addressed by this Evaluation is the Feedwater System. All other affected SSCs are "screened out" by the PAD. These are:

- Installation of instrument cables through secondary containment penetrations.
- Installation of a new panel with new components in the Normal Switchgear Building.
- Additional heat loads in the Containment. NOT DISCUSSED IN THE PAD
- Additional heat loads in the Normal Switchgear Building. NOT DISCUSSED IN THE PAD
- Additional load on the Nonsafety DC Power System.

Existing Level 8 Trips

It should be noted that there are multiple level 8 signals utilized by various systems, and that these level 8 signals are independent; i.e., each system utilizes independent level transmitters. None of the components associated with "MSO level 8," including the level transmitters, trip units and trip coils, are utilized by the "level 8" signals described in the Licensing Basis Documents. Therefore, the "MSO level 8" signal utilized by the proposed MSO Feedwater pump trip is not associated with the "level 8" that:

- Closes the steam supply valve (E51-MOVF045) to the RCIC turbine (UFSAR 7.4.1.1).
- Causes rapid insertion of control rods (scram) (UFSAR 7.2.1.1(4)).
- Causes turbine trip (UFSAR 10.2.6.1).
- Closes the HPCS injection valve (MOVF004) (UFSAR 7.3.1.1.1.1).
- Provides Control Room RPV level indication or alarms (UFSAR 7.7.1.3).

The level transmitters utilized for the proposed change are not associated with any existing trips. This includes the Feedwater Control System RPV water level differential water pressure transmitters which are used to control RPV water level and Feedwater Flow (UFSAR 7.7.1.3(1 & 3)), and the level transmitters that initiate RCIC flow to the RPV upon low (level 2) water level.

Existing Feedwater Pump Trips

The Licensing Basis Documents describe several conditions which will create a loss of Feedwater. From UFSAR Section 15.2.7.1.1:

> *A loss of feedwater flow could occur from pump failures, feedwater controller failures, operator errors, or reactor system variables such as high vessel water*

level (L8) trip signal.

UFSAR Section 10.4.7.5.1 expands on the causes of Feedwater pump trip.

Pushbutton controls are provided in the main control room for manual operation of each reactor feed pump. The following conditions automatically trip a running feed pump:

1. *Pump suction extreme low pressure (two out of three logic)*
2. *Reactor vessel high water level (two out of three logic)*
3. *Lube oil extreme low pressure (after time delay)*
4. *Speed increaser lube oil extreme low pressure*
5. *Bus undervoltage trip*
6. *Electrical protection trip.*

MSO Level 8

"MSO level 8" gets its name due to the fact that it does not have the same nominal setpoint as the other level 8 setpoints[2]. It is set higher than the other level 8 setpoints to prevent its actuation before the other level 8 setpoints. This prevents the new circuit from becoming the initiator of a plant transient (Loss of Feedwater) and it ensures that the other level 8 trips perform their design functions with no influence (such as changed initial conditions) from the MSO level 8.

The range between existing normal level 8 nominal setpoints and the new MSO level 8 setpoint is based on the accuracy of the respective trip circuits. Setpoint calculation G13.18.6.1.C33-001 demonstrates that approximately 9 inches of offset will prevent the two trip setpoints from overlapping. In addition, calculation G13.18.14.0-208 shows that, if the MSO level 8 is needed to shut down the feedwater pumps, there is still sufficient margin between the pump shutdown level and the main steam lines to prevent main steam line flooding and to protect RCIC functionality. Within calculation G13.18.14.0-208 a bounding value of 2 seconds is assumed for the logic delay from the time the actual reactor water level exceeds the process safety limit to the time the power is removed from the feedwater pump motors. Calculation G13.18.14.0-209 confirms that this 2 second delay time is conservative.

MSO Feedwater Trip

This "MSO" Feedwater pump trip function serves as a backup to the level 8 trip which comes from the Control Room. The function of the level 8 trip is to prevent the RPV from overfilling and disabling RCIC function. This function of the RCIC System is described in Technical Specification Bases 3.3.5.2(2):

> *High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal closes the RCIC steam supply valve to prevent overflow into the main steam lines (MSLs).*
>
> *Reactor Vessel Water Level-High, Level 8 signals for RCIC are initiated from two level transmitters from the narrow range water level measurement instrumentation, which sense the difference between the pressure due to a constant column of*

[2] The other level 8 nominal setpoints may not all be the exact same value, but they are very close to each other at about 51" above water level instrument zero. Alternate level 8 is set at a nominal trip setpoint of 59.9".

water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level-High, Level 8 Allowable Value is high enough to preclude isolating the injection valve of the RCIC during normal operation, yet low enough to trip the RCIC System prior to water overflowing into the MSLs.

This is elaborated on by UFSAR Section 7.4.1.1, Reactor Core Isolation Cooling (RCIC) System:

In addition, the steam supply valve MO F045 is closed on reactor vessel high water level (trip level 8). This permits an automatic restart of the RCIC system if the reactor water level drops to level 2.

Based on the Licensing Basis Document descriptions of level 8 trips and feedwater trips it should be noted that there are no new functions provided by the MSO feedwater trip. The feedwater pumps are already provided with a level 8 trip. The MSO trip simply provides a backup for this existing function.

If the water level in the vessel reaches the normal level 8 trip setpoint, the plant <u>will</u> shut down. This is because the normal level 8 trip signals also create a reactor scram, feedwater pump trip, and a turbine trip. Therefore, proper functioning of the MSO level 8 trip will not create a new plant transient or increase the frequency of an existing analyzed plant transient.

Relationship with Other Accidents

A trip of two recirculation pumps (UFSAR 15.3.1.3.3.2), a fast closure of two recirculation valves (UFSAR 15.3.2.4.2) and a recirculation pump shaft break (UFSAR 15.3.4) creates water level surge which is expected to shut down the feedwater pumps on high water level. It is also expected that the high water level will scram the reactor and shut down the main turbine.

However, the MSO feedwater trip is required only if there is a fire in the control room and the main feedwater trips are disabled. In this situation the Operators are assumed to manually scram the reactor. The initiating event for a plant transient that requires the use of the MSO feedwater pump trip circuit is the manual scram, not equipment malfunction.

If there is no fire in the control room then reactor scram for these accidents occurs on normal level 8, as does the feedwater trip. Therefore, the UFSAR accident analyses for both of these accidents are unchanged.

Recirculation Pump Seizure (UFSAR 15.3.3) assumes that the level 8 trip works properly since it makes the event more severe.

These accidents are highlighted because they specifically refer to level 8 as the cause of feedwater pump trip. However, the event for which MSO level 8 feedwater pump trip is designed begins with a fire in the control room, disabled level 8 feedwater pump trips, and a manual scram. No other licensing basis accident scenario is initiated with this sequence of events.

For a fire outside the main control room that impacts the new MSO level 8 trip circuits, multiple hot shorts could create a fire induced spurious trip signal if the multiple hot shorts met the two-out-of-two logic. However, the plant's response to a major fire outside the Main Control Room requires that the reactor feed pumps be tripped if the fire occurs in fire zone AB-10 in accordance with AOP-0052 "Fire Outside the Main Control Room in Areas Containing Safety Related Equipment", so this spurious tripping would have no adverse impact on the plant's ability to achieve and maintain post-fire safe shutdown.

Summary of Evaluation:

The proposed activity adds an additional trip circuit designed to trip the feedwater pumps on a high RPV water level signal. The trip signal is derived from the new narrow range level transmitters. The level signal will be sent through trip units that will be calibrated to create an

MSO level 8 signal. Both trip units must produce an MSO level 8 signal to trip the feedwater pumps ("AND" logic).

MSO level 8 gets its name due to the fact that it does not have the same nominal setpoint as the other level 8 setpoints[3]. It is set higher than the other level 8 setpoints to prevent its actuation before the other level 8 setpoints. This prevents the new circuit from becoming the initiator of a plant transient (Loss of Feedwater) and it ensures that the other level 8 trips perform their design functions with no influence (such as changed initial conditions) from the MSO level 8.

The range between existing normal level 8 nominal setpoints and the new MSO level 8 setpoint is based on the accuracy of the respective trip circuits. Setpoint calculation G13.18.6.1.C33-001, Reactor Vessel Water Level – High, MSO Reactor Level 8, demonstrates that approximately 9 inches of offset will prevent the two trip setpoints from overlapping. In addition, calculation G13.18.14.0-208 shows that, if the MSO level 8 is needed to shut down the feedwater pumps, there is still sufficient margin between the pump shutdown level and the main steam lines to prevent main steam line flooding and to protect RCIC functionality.

Works as Designed

The sole purpose for the MSO level 8 feedwater pump trip is to preserve RCIC functionality in the event of a fire in the control room that disables the feedwater pump trip on high RPV water level. In this event all other level 8 trips are unaffected and work as designed. Normally the feedwater pumps would be tripped at the same time as the other level 8 trips. There will be a time delay between the normal level 8 trip and the MSO level 8 trip. The length of this delay is based on the trip "offset" and the RPV fill rate. Note that the RCIC steam supply valve (E51-MOVF045) will close on a normal level 8 signal because its level transmitters are not associated with the feedwater level transmitters.

[3] The other level 8 nominal setpoints may not all be the exact same value, but they are very close to each other.

Does Not Work as Designed

If the MSO level 8 feedwater pump trip circuit does not work as designed then the main steam lines will flood – but only if the existing level 8 feedwater pump trip system malfunctions or is disabled. Since the MSO level 8 trip is a backup to the existing level 8 trip, failure to work properly does not change the frequency or consequences of reactor vessel overfill.

UFSAR Chapter 15.5, Increase in Reactor Coolant Inventory, was considered as an event which may be effected by a failure of the new MSO level 8 trip to operate properly. However, this event occurs due to an inadvertent High Pressure Core Spray (HPCS) injection, with fire in the main control room not a consideration. In addition, the water level in the RPV for this event does not reach normal level 8. Therefore, MSO level 8 failure to work as designed does not adversely affect an Increase in Reactor Coolant Inventory event.

Spurious Actuation

As demonstrated by the Failure Mode and Effects Analysis (FMEA), a spurious pump trip is a very unlikely event. A thorough review of Operating Experience has only shown one incident that could cause a spurious trip of the Feedwater Pumps. This incident was caused by a slow oil leak within the Rosemount 1153 transmitter sensing cell. A slow leak in the transmitter sensing cell can cause a drift in either direction on the order of ¼ percent or more per month. A drift in the increasing direction could eventually result in a spurious trip of the Reactor Feedwater Pumps. This issue was first identified in CR90-1103 and was addressed in SEN 57 as well as CR-RBS-2013-03661. The defective gaskets have been replaced. Any single failure mechanism from the level transmitters up to and including the Rosemount trip cards will not generate a trip signal due the two-out-of-two logic. For this reason, the new panel will contain visual indication of a half trip signal. This will allow the half trip to be diagnosed and reset.

The Rosemount trip cards will be powered by two Nuclear Logistics, Inc. (NLI) power supplies. Two potential issues with this power supply configuration are as follows:

1. Voltage spikes on the input to the power supplies could cause both of the power supplies to trip off line or could propagate a spike through the power supplies that could initiate a spurious trip on the 4-20 mA circuit associated with the trip units.
2. Voltage spikes generated within the individual power supplies themselves could generate a spurious trip on one of the 4-20 mA circuits at a time.

Although they are not required to be seismic I and safety-related, the NLI power supply will be procured in accordance with River Bend Station – Unit 1, Entergy Operations, Inc. Safety Related (QClass-1) Specification Number 244.513, "Power Supply Assemblies". Section 4.2, "Tests" states as follows:

4.2.1 The Power Supply Assemblies shall demonstrate their ability to withstand repeated inductive spikes on the DC Input and Output lines generated by de-energization of GE Type HFA DC relays without suppression for 24 hours at a repeated rate of at least two (2) cycles per minute. Separate Input and Output Tests are to be carried out with lead minimum cable lengths of two (2) ft. between the relay and the input and output of the test assembly. The units shall operate without tripping any internal protection circuits or damage.

4.2.4 Verify the Power Supply Assemblies are sufficiently filtered to prevent uninterrupted operation of the Rosemount 710 Trip Units.

To ensure that one voltage spike cannot create a trip of both trip units at one time the design provides two different 125 VDC supplies and two different 24 VDC power supplies. This is a more positive control than the capacitor and surge arrestor placed on the power supply's input and the surge arrestor placed on the power supply's outputs. Voltage spikes on the outputs have been an issue with past battery chargers. However the newer Exide battery chargers have not exhibited this same propensity for spikes.

For a fire in the Normal Switchgear Building, multiple hot shorts could result in a spurious (fire-induced) trip signal. However, the pumps are NOT credited for post-fire safe shutdown, and Criterion 240.201a requires that main feed be terminated. Thus, a spurious fire-induced trip of the reactor feed pumps would have no adverse impact on the plant's ability to achieve and maintain safe shutdown.

Is the validity of this Evaluation dependent on any other change? ☐ **Yes** ☒ **No**

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ Yes ☒ No

Preparer: F. Bivins Calhoun III / See AS / ENERCON / Mechanical Design / 2/26/16
Name (print) / Signature / Company / Department / Date

Reviewer: Jorge Merchan / See AS / ENERCON / Electrical Design / 2/26/16
Name (print) / Signature / Company / Department / Date

OSRC: [illegible] / Troy Dean Burnett / 2/26/16
Chairman's Name (print) / Signature / Date

16-04 / EN-2016-01
OSRC Meeting #

II. 50.59 EVALUATION **Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.**		☐ ☒	Yes No
Does the proposed Change:			
1.	Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR?	☐ ☒	Yes No
	BASIS: The only event frequency that the proposed activity can affect is Loss of Feedwater Flow (UFSAR 15.2.7). While Feedwater pump trip on high RPV water level is the desired outcome, the new trip circuit does introduce the possibility of a spurious trip. The causes of loss of feedwater flow are described in UFSAR Section 15.2.7.1.1: *A loss of feedwater flow could occur from pump failures, feedwater controller failures, operator errors, or reactor system variables such as high vessel water level (L8) trip signal.* UFSAR Section 10.4.7.5.1 expands on the causes of Feedwater pump trip. *Pushbutton controls are provided in the main control room for manual operation of each reactor feed pump. The following conditions automatically trip a running feed pump:* *1. Pump suction extreme low pressure (two out of three logic)* *2. Reactor vessel high water level (two out of three logic)* *3. Lube oil extreme low pressure (after time delay)* *4. Speed increaser lube oil extreme low pressure* *5. Bus undervoltage trip* *6. Electrical protection trip.* The Loss of Feedwater event is categorized as an incident of moderate frequency (1 in 20 years or frequency of 0.05) (USAR 15.2.7.1.2). The increased frequency is negligible for a Loss of Feedwater event based on evaluated 95% confidence bands for existing Level 8 Trips and new MSO Level 8 Trip (Calculation G13.18.14.0-208). An incident of moderate frequency is defined by UFSAR Section 15.0.3.1(1):		

Incidents of moderate frequency - These may occur during a calendar year to once per 20 yr for a particular plant. This event is referred to as an "anticipated (expected) operational transient."

The UFSAR already acknowledges that a loss of feedwater may be caused by high vessel water level (level 8). MSO level 8 is set higher than the existing level 8 trip such that it will be effective only if the primary level 8 trip fails. Therefore, during normal operation, the MSO level 8 trip circuit is not a new accident initiator.

MSO level 8 gets its name due to the fact that it does not have the same nominal setpoint as the other level 8 setpoints. It is set higher than the other level 8 setpoints to prevent its actuation before the other level 8 setpoints. This prevents the new circuit from becoming the initiator of a plant transient (Loss of Feedwater) and it ensures that the other level 8 trips perform their design functions with no influence (such as changed initial conditions) from the MSO level 8.

The range between existing normal level 8 nominal setpoints and the new MSO level 8 setpoint is based on the accuracy of the respective trip circuits. Setpoint calculation G13.18.6.1.C33-001, Reactor Vessel Water Level – High, MSO Reactor Level 8, demonstrates that approximately 9 inches of offset will prevent the two trip setpoints from overlapping. In addition, calculation G13.18.14.0-208 shows that, if the MSO level 8 is needed to shut down the feedwater pumps, there is still sufficient margin between the pump shutdown level and the main steam lines to prevent main steam line flooding and to protect RCIC functionality.

The modification adds an additional trip circuit designed to trip all feedwater pumps given the combination of a reactor water level 8 (MSO level 8) signal derived from new dedicated RPV level transmitters. The same level MSO 8 signal is used for all three of the feedwater pumps. Spurious actuation of this trip circuit would result in a loss of all feedwater. Other potential spurious losses of feedwater include 2 out of 3 level 8 signals which trip the pump breakers, trips associated with protection of the feed pump motors and power supplies, and trips associated with low lube oil pressure. In addition, pump failures, feedwater controller failures, operator errors or reactor system variables (high level) can cause a loss of feedwater.

This modification involves the addition of completely new and independent circuitry intended to generate a feed water pump trip signal. Thus, this circuitry, as is true with any such addition, is theoretically capable of inducing an unwanted feedwater pump trip.

The FMEA developed in support of this effort has assessed this potential and has concluded that the design proposed by EC 59951 has satisfied the original criteria placed on this design as follows:

- This design change is compliant with the best known design practices, including hardened and independent power supplies, independent 125 VDC supplied circuits, appropriate fire protection

techniques, and separate transmitters providing input to a 2-out-of-2 logic circuit.

- This design will not jeopardize the health and the safety of the public in that the appropriate selectivity of the 2-out-of-2 logic trip channels have been properly designed.
- This design Change does not constitute a more than minimal increase in the probability of an accident.

This conclusion is the result of assessing the high level of quality and independence of the design elements associated with this effort. Specifically, the features that contribute this high level of reliability are thorough reviews of the Operational Experience (OE) associated with the four components which constitute the new trip circuit:

- Rosemount 1153DB4PA Differential Pressure Transmitters
- NUS (Rosemount) 710DU Trip Units
- NLI (VICOR VI-200) Power Supplies
- Exide SCFR130-3-300 Battery Chargers

The OE for these components, as well as additional design features, were evaluated by the FMEA:

1. Rosemount 1153DB4PA Differential Pressure Transmitters
   a. Slow leak or loss of oil in the transmitter sensing cell. This could cause a leak such that a spurious one-half (1/2) trip of one feedwater pump could occur. This leaking condition is a slow process that has not been occurring since the original bad batch of seals have been replaced.
   b. Particles in the sensor cell could cause an instantaneous shift of the output to off scale high. This has not been observed since the original transmitters were replaced in 1984. Manufacturing process changes at Rosemount have eliminated particles in the transmitter sensor cells.
2. NUS (Rosemount) NUS 710DU
   a. Failure of a setpoint adjustment potentiometer resulted in a ½ pump trip logic (one trip unit could trip). It is unlikely that both trip units would fail at the same time.
3. NLI (VICOR VI-200) Power Supplies
   a. No specific anomalies associated with these power supplies since they were first utilized at River Bend Station with MR-95-0023. These power supplies were procured and installed in accordance with River Bend Unit 1, ENTERGY Operations, Inc., Safety Related (QClass-1) Specification Number 244.513, "Power Supply Assemblies," Rev. 0, dated 08/02/1995. These power supplies have been specifically fitted with a capacitor and a surge arrestor on the input side of the power supply and a surge arrestor on the output side of the power supply. These components have been added to harden the power supply against transients specifically generated by the inductive voltage spikes associated with GE Type HFA relays.
4. Exide SCFR130-3-300
   a. Battery auto-tripped on high voltage while adjusting equalizing charge voltage caused by a faulty equalize voltage adjustment potentiometer. This occurred on an old battery charger that has since been replaced. The new battery charger has not

demonstrated this failure mode since initial installation.

b. In the past various electronically controlled power components with Silicon Controlled Rectifiers (SCRs) have been known to create erratic voltage and current signals as a result of miss-firings of the SCRs. This was a problem associated with previously utilized Exide Battery Chargers. However, this anomaly has not been known to occur with the presently installed chargers.

Additional concerns associated with the initial design change was that a potential exists for one component failure to cause perturbation of both trip units and/or both transmitters. The initial design provided one power supply apparatus with two power supply outputs. The initial design also provided only one power source from one disconnect from the 125 VDC power distribution panel. One of the assumed potential failure modes was that of a Voltage Surge from a battery charger or internal to a power supply could cause a spurious actuation of both trip units from one spike. As a result of this presumption it was determined that independence of the 125 VDC power sources and 24 VDC power supplies would preclude a single spurious actuation from tripping the reactor feedwater pumps.

In addition to specific components being evaluated, the configuration of the cable routing was also evaluated. The cables specifically associated with the MSO level eight trip are not energized until the trip signal is initiated. In order to spuriously actuate the auxiliary trip circuits for more than one reactor feed pump, two inter-cable ground fault equivalent hot shorts in conduit OR two inter-cable hot shorts within cable tray would be required. Per Regulatory Guide 1.189 (as supplemented by NUREG CR-7150), such fire scenarios are not credible. If a fire does exist, then the trip of the reactor feed pumps is consistent with what is required for a response to a fire anywhere along this designed route. Thus, introduction of the new cables will not create any new condition that has not already been evaluated. That is, the currently credited means of securing reactor feed water should still be available; any postulated cable damage could only trip the reactor feed pumps, which places them in the required safe shutdown position.

Modifications, such as this one, which involve a vanishing small, theoretical deleterious impact are specifically addressed NEI 96-07, Revision 1, Section 4.3.1 (accidents), Example 1. Specifically, the definition of a "negligible effect" is defined as (emphasis added):

> The proposed activity has a negligible effect on the frequency of occurrence of an accident. A negligible effect on the frequency of occurrence of an accident exists when the **change in frequency is so small** or the uncertainties in determining whether a change in frequency has occurred are such that it **cannot be reasonably concluded that the frequency has actually changed** (i.e., there is no clear trend toward increasing the frequency).

Therefore, the proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

2.	Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?	☐ ☒	Yes No

BASIS:

This change installs an additional trip circuit for the feedwater pumps. All applicable design and functional requirements, applicable codes and standards will be met as part of the design. As such, the modification would not cause more than a minimal increase in the likelihood of a malfunction. The change in likelihood of a malfunction (spurious trip of operating feedwater pump) is estimated to be no more than a minimal increase since there are already other circuits that will trip a pump. Therefore, the increase in likelihood of a malfunction is minimal.

The causes of loss of feedwater flow are described in UFSAR Section 15.2.7.1.1:

> *A loss of feedwater flow could occur from pump failures, feedwater controller failures, operator errors, or reactor system variables such as high vessel water level (L8) trip signal.*

Since a level 8 trip is already anticipated (and desired), adding an MSO level 8 trip to back up the existing trip does not introduce a new failure mode.

The proposed change does not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. The proposed change does not degrade the safety function of any safety related equipment or equipment important to safety which might mitigate malfunction of equipment. The proposed change does not degrade the performance of, or increase the challenges to, any safety system assumed to function in the UFSAR accident analyses.

This modification involves the addition of completely new and independent circuitry intended to generate a feed water pump trip signal. Thus, this circuitry, as is true with any such addition, is theoretically capable of inducing an unwanted feedwater pump trip. All devices are analog so there are no digital components.

MSO level 8 gets its name due to the fact that it does not have the same nominal setpoint as the other level 8 setpoints. It is set higher than the other level 8 setpoints to prevent its actuation before the other level 8 setpoints. This prevents the new circuit from becoming the initiator of a plant transient (Loss of Feedwater) and it ensures that the other level 8 trips perform their design functions with no influence (such as changed initial conditions) from the MSO level 8.

The range between existing normal level 8 nominal setpoints and the new MSO level 8 setpoint is based on the accuracy of the respective trip circuits. Setpoint calculation G13.18.6.1.C33-001, Reactor Vessel Water Level – High, MSO Reactor Level 8, demonstrates that approximately 9 inches of offset will prevent the two trip setpoints from overlapping. In addition, calculation G13.18.14.0-208 shows that, if the MSO

level 8 is needed to shut down the feedwater pumps, there is still sufficient margin between the pump shutdown level and the main steam lines to prevent main steam line flooding and to protect RCIC functionality.

The FMEA developed in support of this effort has assessed this potential and has concluded that the final design proposed by EC 59951 has satisfied the original criteria placed on this design as follows:

- This design change is compliant with the best known design practices, including hardened and independent power supplies, independent 125 VDC supplied circuits, appropriate fire protection techniques, and two independent transmitters providing input to two out of two trip logic.
- This design will not jeopardize the health and the safety of the public in that the appropriate selectivity of two out of two trip logic channels have been properly designed.
- This design Change does not constitute a more than minimal increase in the probability of an important-to-safety SSC malfunction.

This conclusion is the result of assessing the high level of quality and independence of the design elements associated with this effort. Specifically, the features that contribute this high level of reliability are thorough reviews of the Operational Experience (OE) associated with the four components which constitute the new trip circuit:

- Rosemount 1153DB4PA Differential Pressure Transmitters
- NUS (Rosemount) 710DU Trip Units
- NLI (VICOR VI-200) Power Supplies
- Exide SCFR130-3-300 Battery Chargers

The OE for these components, as well as additional design features, were evaluated by the FMEA:

1. Rosemount 1153DB4PA Differential Pressure Transmitters
    a. Slow leak or loss of oil in the transmitter sensing cell. This could cause a leak such that a spurious one-half (1/2) trip of one feedwater pump could occur. This leaking condition is a slow process that has not been occurring since the original bad batch of seals have been replaced.
    b. Particles in the sensor cell could cause an instantaneous shift of the output to off scale high. This has not been observed since the original transmitters were replaced in 1984. Manufacturing process changes at Rosemount have eliminated particles in the transmitter sensor cells.
2. NUS (Rosemount) NUS 710DU
    a. Failure of a setpoint adjustment potentiometer resulted in a ½ pump trip logic (one trip unit could trip). It is unlikely that both trip units would fail at the same time.
3. NLI (VICOR VI-200) Power Supplies
    a. No specific anomalies associated with these power supplies since they were first utilized at River Bend Station with MR-95-0023. These power supplies were procured and installed in accordance with River Bend Unit 1, ENTERGY Operations, Inc., Safety Related (QClass-1) Specification Number 244.513, "Power Supply

Assemblies," Rev. 0, dated 08/02/1995. These power supplies have been specifically fitted with a capacitor and a surge arrestor on the input side of the power supply and a surge arrestor on the output side of the power supply. These components have been added to harden the power supply against transients specifically generated by the inductive voltage spikes associated with GE Type HFA relays.

4. Exide SCFR130-3-300

   a. Battery auto-tripped on high voltage while adjusting equalizing charge voltage caused by a faulty equalize voltage adjustment potentiometer. This occurred on an old battery charger that has since been replaced. The new battery charger has not demonstrated this failure mode since initial installation.

   b. In the past various electronically controlled power components with Silicon Controlled Rectifiers (SCRs) have been known to create erratic voltage and current signals as a result of miss-firings of the SCRs. This was a problem associated with previously utilized Exide Battery Chargers. However, this anomaly has not been known to occur with the presently installed chargers.

Additional concerns associated with the initial design change was that a potential exists for one component failure to cause perturbation of both trip units and/or both transmitters. The initial design provided one power supply apparatus with two power supply outputs. The initial design also provided only one power source from one disconnect from the 125 VDC power distribution panel. One of the assumed potential failure modes was that of a Voltage Surge from a battery charger or internal to a power supply could cause a spurious actuation of both trip units from one spike. As a result of this presumption it was determined that the independence of the 125 VDC power sources and 24 VDC power supplies would preclude a single spurious actuation from tripping the reactor feedwater pumps.

In addition to specific components being evaluated, the configuration of the cable routing was also evaluated. The cables specifically associated with the MSO level eight trip are not energized until the trip signal is initiated. In order to spuriously actuate the auxiliary trip circuits for more than one reactor feed pump, two inter-cable ground fault equivalent hot shorts in conduit OR two inter-cable hot shorts within cable tray would be required. Per Regulatory Guide 1.189 (as supplemented by NUREG CR-7150), such fire scenarios are not credible. If a fire does exist, then the trip of the reactor feed pumps is consistent with what is required for a response to a fire anywhere along this designed route. Thus, introduction of the new cables will not create any new condition that has not already been evaluated. That is, the currently credited means of securing reactor feed water should still be available; any postulated cable damage could only trip the reactor feed pumps, which places them in the required safe shutdown position.

Modifications, such as this one, which involve installing additional equipment or devices are specifically addressed NEI 96-07, Revision 1, Section 4.3.2 (malfunctions), Example 1:

> The change involves installing additional equipment or devices (e.g., cabling, manual valves, protective features) provided all applicable design and functional requirements (including applicable codes, standards, etc.) continue to be met. For example, adding protective devices to breakers or installing an additional drain line (with appropriate isolation capability) would not cause more than a minimal increase in the likelihood of malfunction.

Example 6 from the same section is also applicable in that the change does <u>not</u> reduce system/equipment

	redundancy, diversity, separation or independence. Therefore, the proposed activities will not increase the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.		
3.	Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR?	☐ ☒	Yes No
	BASIS: This change installs an additional level 8 trip circuit for the feedwater pumps. The new circuit is used for nothing other than tripping the feedwater pumps upon high water level in the RPV. Loss of Feedwater is an anticipated event and the proposed activity will not change the consequences of loss of Feedwater. In addition, the Feedwater pumps are non-Safety Related and are not required for safe shutdown. Nor will the new trip circuit interfere with or affect those systems which are required for safe shutdown and/or mitigating the consequences of an accident. The "Loss of Feedwater" incident has no radiological consequences, as indicated by UFSAR Section 15.2.7.5: *The consequences of this event do not result in any fuel failure. Therefore, no analysis of the radiological consequences is required.* The new trip circuit does not change any actions or sequence of events after the initial pump trip, so the UFSAR conclusion regarding radiological consequence is unchanged. Therefore, the proposed activity will not result in an increase in the consequences of any accident previously evaluated in the UFSAR.		
4.	Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?	☐ ☒	Yes No

BASIS:

This change installs an additional level 8 trip circuit for the feedwater pumps. The new circuit is used for nothing other than tripping the feedwater pumps upon high water level in the RPV. Loss of Feedwater is an anticipated event and the proposed activity will not change the consequences of loss of Feedwater. In addition, the Feedwater pumps are non-Safety Related and are not required for safe shutdown. Nor will the new trip circuit interfere with or affect those systems which are required for safe shutdown and/or mitigating the consequences of an accident.

The "Loss of Feedwater" incident has no radiological consequences, as indicated by UFSAR Section 15.2.7.5:

> *The consequences of this event do not result in any fuel failure. Therefore, no analysis of the radiological consequences is required.*

The new trip circuit does not change any actions or sequence of events after the initial pump trip, so the UFSAR conclusion regarding radiological consequence is unchanged.

Therefore, the proposed change will not result in more than a minimal increase in the consequences of a malfunction of SSC important to safety previously evaluated in the UFSAR.

5.	Create a possibility for an accident of a different type than any previously evaluated in the UFSAR?	☐ ☒	Yes No
	BASIS: This design change improves the ability to be able to respond to a Fire in the Main Control Room coupled with a SCRAM that can cause the water levels in the reactor to rise above level 8. This change bypasses the main control room to ensure that the reactor feed pumps will trip on level 8 if multiple spurious operations and circuit failures have caused the reactor feed pump control power to disable the primary control fuses. There are no new sources of accident initiation created by the proposed activity. Loss of feedwater on high RPV water level is already acknowledged by the UFSAR. The existing Feedwater pump trip creates no action other than to trip the pumps. The MSO trip setpoint is arranged such that it will actuate only after the existing level 8 trips – thus ensuring no other required actions created by the existing level 8 trip will be bypassed. Calculation G13.18.6.1.C33-001 demonstrates that the MSO level 8 trip setpoint is sufficiently above the existing level 8 setpoint such that there is a negligible chance for setpoint overlap that may occur due to instrument inaccuracies. Calculation G13.18.14.0-208 demonstrates that MSO level 8 trip setpoint will prevent the main steam lines from flooding, thus protecting RCIC functionality. No new Operator actions are required to implement a level 8 trip. The MSO level 8 trip, just as the existing level 8 trip, is automatic and requires no Operator actions. Nor does the MSO trip circuit provide any new alarms or level indication which could influence Operator actions. Therefore, the proposed changes do not create the possibility of an accident of a different type than previously evaluated in the UFSAR.		

6.	Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR?	☐ ☒	Yes No
	BASIS: This design change improves the ability to be able to respond to a Fire in the Main Control Room coupled with a SCRAM that can cause the water levels in the reactor to rise above level 8. This change bypasses the main control room to ensure that the reactor feed pumps will trip on level 8 if multiple spurious operations and circuit failures have caused the reactor feed pump control power to disable the primary control fuses. The existing Feedwater pump trip creates no action other than to trip the pumps. The MSO trip setpoint is arranged such that it will actuate only after the existing level 8 trips – thus ensuring no other required actions created by the existing level 8 trip will be bypassed. Calculation G13.18.6.1.C33-001 demonstrates that the MSO level 8 trip setpoint is sufficiently above the existing level 8 setpoint such that there is no chance for setpoint overlap that may occur due to instrument inaccuracies. Calculation G13.18.14.0-208 demonstrates that MSO level 8 trip setpoint will prevent the main steam lines from flooding, thus protecting RCIC functionality. No new Operator actions are required to implement a level 8 trip. The MSO level 8 trip, just as the existing level 8 trip, is automatic and requires no Operator actions. Nor does the MSO trip circuit provide any new alarms or level indication. Therefore, the proposed changes will not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR.		
7.	Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered?	☐ ☒	Yes No

	BASIS: This design change utilizes known, reliable components to provide an alternate means of ensuring the reactor feed pumps can be tripped on a level 8 in the reactor even with a fire in the control room that creates multiple spurious actuations in the control room that causes 125 VDC control power to be lost to the reactor feed pumps. Present designs and procedures are dependent on operator actions to provide the alternate means for tripping these pumps. Calculations demonstrate that it only takes approximately thirty-nine (39) seconds for feedwater to progress from a level 8 in the reactor vessel to filling up the main steam lines. This change improves the ability to ensure that damage does not occur. This precludes a potential challenge to the RCIC system by making it possible to automatically trip the reactor feed pumps with a fire in the main control room. RCIC has been credited for maintaining reactor water level in the vessel upon a SCRAM. This control is not viable if the RCIC is made unavailable by flooding the main steam lines. This design change improves the ability to maintain this source of water to the vessel with a fire in the main control room.		
8.	Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses?	☐ ☒	Yes No
	BASIS: The proposed activity is a physical change and does not involve a change to an element of a UFSAR described evaluation methodology, nor does it use an alternate evaluation methodology used in establishing the design basis or used in the safety analysis. The proposed activity does not require the revision of any of the safety analyses. Therefore, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR that is used to establish the design bases or used in the safety analysis.		

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

Project ENTGRB179

Multiple Spurious Operation (MSO)

Alternate Level Eight (8) Feedpump Trips

Due to Fire in the Main Control Room

Failure Mode and Effects Analysis (FMEA)

A Failure Mode and Effects Analysis (FMEA) was performed at a similar level of rigor as was performed for the original feedwater system as described in detail in the USAR on the pertinent details of these engineering changes. During this FMEA four specific components were considered to be the primary components that drive this evaluation as follows:

1. Rosemount 1153DB4PA Differential Pressure Transmitters
2. NUS (Rosemount) 710DU Trip Units
3. NLI (VICOR VI-200) Power Supplies
4. Exide SCFR130-3-300 Battery Chargers

The Operability Experience Matrix is provided in Attachment # 1. Some additional discussion is necessary to properly evaluate this operating experience report. The failures we are concerned with are spurious actuations that could cause a preliminary trip of the reactor feedwater pumps. The types of spurious actuations are as follows:

1. Rosemount 1153DB4PA Differential Pressure Transmitters
    a. Slow leak or loss of oil in the transmitter sensing cell. This could cause a leak such that a spurious one-half (1/2) trip of one feedwater pump could occur. This leaking condition is a slow process that has not been occurring since the original bad batch of seals have been replaced.
    b. Particles in the sensor cell could cause an instantaneous shift of the output to off scale high. This has not been observed since the original transmitters were replaced in 1984. Manufacturing process changes at Rosemount have eliminated particles in the transmitter sensor cells.
2. NUS (Rosemount) NUS 710DU
    a. Failure of a setpoint adjustment potentiometer resulted in a ½ pump trip logic (one trip unit could trip). It is unlikely that both trip units would fail at the same time.

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Multiple Spurious Operation (MSO)

Alternate Level Eight (8) Feedpump Trips

Due to Fire in the Main Control Room

Failure Mode and Effects Analysis (FMEA)

3. NLI (VICOR VI-200) Power Supplies
    a. No specific anomalies associated with these power supplies since they were first utilized at River Bend Station with MR-95-0023. These power supplies were procured and installed in accordance with River Bend Unit 1, ENTERGY Operations, Inc., Safety Related (QClass-1) Specification Number 244.513, "Power Supply Assemblies," Rev. 0, dated 08/02/1995. These power supplies have been specifically fitted with a capacitor and a surge arrestor on the input side of the power supply and a surge arrestor on the output side of the power supply. These components have been added to harden the power supply against transients specifically generated by the inductive voltage spikes associated with GE Type HFA relays.
4. Exide SCFR130-3-300
    a. Battery auto-tripped on high voltage while adjusting equalizing charge voltage caused by a faulty equalize voltage adjustment potentiometer. This occurred on an old battery charger that has since been replaced. The new battery charger has not demonstrated this failure mode since initial installation.
    b. In the past various electronically controlled power components with Silicon Controlled Rectifiers (SCRs) have been known to create erratic voltage and current signals as a result of miss-firings of the SCRs. This was a problem associated with previously utilized Exide Battery Chargers. However, this anomaly has not been known to occur with the presently installed chargers.

Additional concerns associated with the initial design change was that a potential exists for one component failure to cause perturbation of both trip units and/or both transmitters. The initial design provided one power supply apparatus with two power supply outputs. The initial design also provided only one power source from one disconnect from the 125 VDC power distribution panel. One of the assumed potential failure modes was that of a Voltage Surge from a battery charger or internal to a power supply could cause a spurious actuation of both trip units from one spike. As a result of this presumption it was determined that independence of the 125 VDC power sources and 24VDC power supplies would preclude a single spurious actuation from tripping the reactor feedwater pumps.

In addition to specific components being evaluated, the configuration of the cable routing was also evaluated. The cables specifically associated with the alternate MSO level eight trip are not energized until the trip signal is initiated. In order to spuriously actuate the auxiliary trip circuits for more than one reactor feed pump, two inter-cable ground fault equivalent hot shorts in conduit OR two inter-cable hot shorts within cable tray would be required. Per Regulatory Guide 1.189 (as supplemented by NUREG CR-7150), such fire scenarios are not credible. If a fire does exist, then the trip of the reactor feed pumps is consistent with what is required for a response to a fire anywhere along this designed route. Thus,

Project ENTGRB179

Multiple Spurious Operation (MSO)

Alternate Level Eight (8) Feedpump Trips

Due to Fire in the Main Control Room

Failure Mode and Effects Analysis (FMEA)

introduction of the new cables will not create any new condition that has not already been evaluated. That is, the currently credited means of securing reactor feed water should still be available; any postulated cable damage could only trip the reactor feed pumps, which places them in the required safe shutdown position.

Conclusions of the FMEA:

From this documented FMEA it is concluded that the final design proposed by EC 59951 has satisfied the original criteria placed on this design as follows:

1. This design change is compliant with theapplicable codes and standards as applied to the original design of the feedwater system as described in the USAR, including hardened and independent power supplies, independent 125 VDC supplied circuits, appropriate fire protection techniques, and two independent transmitters providing input to two out of two trip logic.
2. This design will not has the appropriate selectivity of independent two out of two trip channels that will minimize inadvertent trips of the feedwater pumps while ensuring that a trip will take place upon the vessel level reaching the defined MSO level eight (8) trip on both trip channels.
3. This design Change does not constitute a more than minimal increase of the probability or the consequences of a malfunction. Other portions of this evaluation are covered by the remainder of the attached 50.59 evaluation.

VENDOR DEVICE	EQ COMPONENT TAG	GROUP OR FUNCTION	WHAT HAPPENS	EFFECT ON BALANCE OF SYSTEM	COMMENT	CONCLUSION
Rosemount 1153DB4PA	C33-LTN006A/B	Reactor Pressure Vessel Level	A slow leak/loss of oil may develop in the transmitter sensing cell potentially causing the following symptoms: (1) slow drift in either direction on the order of 1/4 percent or more per month, (2) a deviation from the normal system signal fluctuation that is consistent in only the increasing or decreasing direction, (3) a slow response to transients or failure to follow a transient, (4) a decrease in the root mean square noise level, (5) deviation of one channel from redundant channels.	The slow drift in an increasing direction could eventually result in a spurious trip of the Reactor Feedwater Pumps.	SEN 57, CR-RBS-2013-03661	Not Applicable: Rosemount has since replaced the gasket of the sensor cell to correct the issue; RBS has corrective actions in place to ensure transmitters with serial numbers lower than 500000 (the affected batch) are replaced, identified through receipt inspection, and/or are not installed.
Rosemount 1153DB4PA	C33-LTN006A/B	Reactor Pressure Vessel Level	Newer Rosemount Alphaline Nuclear Pressure Transmitters have a higher current limit than older transmitters. Transmitters that are saturated during normal plant operation should have the Gross Fail High Setpoint adjusted accordingly.	None. During normal plant operation the transmitters are not in a saturated condition.	OE23524	Not Applicable: This event is applicable to transmitters that are saturated during normal plant operation. These transmitters are not installed in a saturated condition application.
Rosemount 1153DB4PA	C33-LTN006A/B	Reactor Pressure Vessel Level	Over torqueing Rosemount 1153 terminal screws could lead to cracked terminal screws and loose/intermittent connections.	Intermittent connections would cause the signal to go to 0 mA. This condition could lead to a failure of the alternate Reactor Feedwater Pump trip scheme. This condition would not cause spurious trips of the Reactor Feedwater Pumps.	O&MR 288-85	Not Applicable: Installers should be sure to torque the screw terminals per Rosemount 1153 instruction manual (5 in-pounds).
Rosemount 1153DB4PA	C33-LTN006A/B	Reactor Pressure Vessel Level	Particles in the sensor cell caused an instantons shift of the output to off scale values. This can lead to transmitters being found of out of tolerance.	A sudden output shift to off scale high would cause a trip of the Reactor Feedwater Pumps.	SEN 336-88	Not Applicable: The event documented in SEN 336-88 occurred in 1984. Rosemount took corrective actions in their manufacturing processes to eliminate particles in sensor cells.
NUS 710DU (Rosemount 710DU)	C33-ESN006A/B	Reactor Pressure Vessel Level 8 Trip	Setpoint adjustment potentiometer (R11) failed and numerous intermittent Reactor Level 8 trip alarms.	Failure of one trip unit would result in 1/2 pump trip logic coming in. Failure of two trip units would result in pump trip.	OE15888	Not Applicable: The failure of the adjustment potentiometer (R11) is considered an anomaly; concurrent failures of this type of multiple trip units is unlikely. At the time of this event there were no other documented failures of this type. A search of OE subsequent to this incident did not result any other additional events documenting a failure of this type.
NUS 710DU (Rosemount 710DU)	C33-ESN006A/B	Reactor Pressure Vessel Level 8 Trip	Capacitor (C25) failed and resulted in loss of input power.	Loss of input power to the trip unit would lead to a failure of the alternate Reactor Feedwater Pump trip scheme. This condition would not cause spurious trips of the Reactor Feedwater Pumps.	OE307423	Not Applicable: The failure of capacitor (C25) would not result in the spurious trips of the Reactor Feedwater Pumps. No other events were found concerning capacitor (C25) failure during the OE search.
NUS 710DU (Rosemount 710DU)	C33-ESN006A/B	Reactor Pressure Vessel Level 8 Trip	Noise on the 24VDC power supply for the 710DU trip systems is coupled back to input while testing channels using Transmation 1040 calibrator in transmitter simulate mode to input a 4-20 mADC signal.	None. The event occurred during calibration of the trip unit using a Transmation 1040.	OE14187	Not Applicable: This is a testing issue not an operational issue. This condition is present during calibration using a Transmation 1040 calibrator. Use an alternate (Fluke 702) calibrator to calibrate the trip units.
NUS 710DU (Rosemount 710DU)	C33-ESN006A/B	Reactor Pressure Vessel Level 8 Trip	Failure of resistor (R59) in the analog meter section of the Master Trip Unit circuit caused a downscale reading.	The analog meter would read incorrectly. The trip function of the trip unit is not affected.	OE35963, CR-RBS-2012-04265, CR-RBS-2012-04267	Not Applicable: This failure would not cause spurious trips of the Reactor Feedwater Pumps. This failure does not affect the trip function of the master trip unit.
NUS 710DU (Rosemount 710DU)	C33-ESN006A/B	Reactor Pressure Vessel Level 8 Trip	Direct exposure to steam and hot moisture resulted damage to and failure of trip unit circuit card.	Failure of the trip card would lead to a failure of the alternate Reactor Feedwater Pump trip scheme. This condition would not cause spurious trips of the Reactor Feedwater Pumps.	OE36287	Not Applicable: This failure would not cause spurious trips of the Reactor Feedwater Pumps. The trip unit is not in a location where it will come into direct contact with steam and hot moisture.
NUS 710DU (Rosemount 710DU)	C33-ESN006A/B	Reactor Pressure Vessel Level 8 Trip	Trip units placed in locations that may be inside accident level radiation fields may not reliably trip at specified trip point.	None. The trip units are installed in the Normal Switchgear Building	CR-RBS-2013-06170	Not Applicable: The trip units are located in a mild environment.
Exide SCFR130-3-300	BYS-CHGR1B	Battery Charger	Battery auto-tripped on high voltage while adjusting equalizing charge voltage caused by a faulty equalize voltage adjustment potentiometer.	Voltage could lead to initiation of Reactor Feedwater Pumps trip.	OE301303	Not Applicable: The apparent cause of the failure was a failure of the station to implement refurbishment PMs for battery chargers in a timely manner.

I. OVERVIEW / SIGNATURES[1]

Facility: River Bend Station **Evaluation #2016-002 / Rev. #:** 0

Proposed Change / Document: EC No. 54958

Description of Change: The existing General Electric (GE) Mark II turbine electro-hydraulic control (EHC) system at River Bend Station (RBS) is obsolete, and has a number of disadvantages compared to modern digital EHC (DEHC) system designs.

The design objective of this EC is to install a modern digital turbine control system (Westinghouse Ovation®) to replace the current analog C85 Steam Bypass and Pressure Control System and the Mark II Turbine Control system, while making use of previously installed hardware, as feasible or dictated by contract terms. General design objectives include the following:

- Replace the current control system to eliminate single point vulnerabilities, address component obsolescence, provide future expandability for additional system interfaces, improve maintenance and diagnostic capabilities, and provide operational improvements
- Enhance the reliability of the control system via redundant control systems, redundant and diverse overspeed protection systems, redundant control of bypass valves, stop valves, control valves, and combined intercept valves
- Provide triple redundant pressure transmitters in place of pressure switches for critical turbine functions.
- Provide the ability to perform automatic pressure rate control for reactor heatup and cooldown
- Provide test compensation logic during control valve testing to minimize fluctuations in reactor and main steam line pressure
- Provide a direct power load unbalance (PLU) turbine trip

The new DEHC turbine control system (TCS) is modeled based on parameters of the existing C85 Steam Bypass and Pressure Regulation System and the Mark II Turbine Control System and will provide the equivalent functions and control schemes as the original pressure and turbine control systems but will include an improved control strategy. The new system does not change the performance requirements of any pump or valve (e.g. stroke time).

The proposed activity also includes:

- Changes to the USAR that revised the control system description but does not change the functional requirements in the USAR. Refer to LBDCR 10.02-020
- A change to operating and maintenance procedures (new and revised) that incorporate the differences in the control system specifics (e.g., hard control stations replaced by soft control graphical interfaces). The procedure changes will not alter the functions of the Pressure Regulation or Turbine-Generator Systems as described in the USAR.
- Vendor Testing of the new DEHC:
  - Software in the Loop Testing (SWIL)
  - Factory Acceptance Test (FAT)
- Site acceptance, post modification, and operability testing. A Site Acceptance Test will be performed prior to installation, during which functional testing will be performed. A Post Modification Test (PMT) will be performed after installation but prior to unit startup to complete calibrations and functional testing of the DEHC system. System testing during power ascension will include performance verifications during unit operation as well as control system tuning. Functional testing will be performed under an Engineering Change Test (ECT) developed to implement requirements of the Post Modification Test Plan (PMTP) for EC-54958.

NOTE: The proposed activity uses Linear Variable Reluctance Transformers (LVRTs) to measure steam valve position indication. However, the nomenclature throughout project documents used the term Linear Variable Differential Transformers (LVDTs). Both components are linear position sensing devices. Therefore, this 10 CFR 50.59 Evaluation form will use the term LVDT for documentation consistency and to avoid any confusion.

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[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

Summary of Evaluation: The following changes associated with the proposed activity were judged to fundamentally alter the existing means of performing or controlling design functions and are reviewed in the attached 50.59 Evaluation:

1. Analog control system to a digital control system since the digital control system may contain different failure modes than the analog system. Reference Failure Modes and Effects Analysis (FMEA) W120-0312.
2. Change from functionally diverse and redundant mechanical and electrical turbine trip mechanisms to a redundant but electrically diverse turbine trip mechanisms. Reference Reliability Analysis and Electronic Overspeed Protection Fault Tree Analysis W120-0346.
3. Conversion from hard controls to soft controls as it involves more than a minimal difference in Human Machine Interface (HMI). Note that the manual turbine trip pushbuttons (new hard controls) located locally at the turbine and remotely in the Main Control Room at control panel H13-P680 are hardwired to the testable dump manifold's (TDM) solenoid valves. The manual trip does NOT get processed by the new control system but only monitored by the system.

This 50.59 Evaluation has determined that the proposed activity does not result in operation of equipment outside the design functions as currently described in the USAR. The upgraded turbine and steam bypass pressure control system (referred to collectively as the Ovation® DEHC) will perform the same functions using the same process inputs (e.g., throttle pressure, turbine speed, generator current, intermediate pressure, discrete inputs) within the same operational limits as previously required for the EHC and C85 Steam Bypass and Pressure Control Systems as described in the Westinghouse Software Hazards Analysis (7231.015-001-001). The malfunctions and accidents currently analyzed in the USAR for the EHC system are therefore bounding for the DEHC system. In addition, the proposed change does NOT create the possibility for an accident or malfunction of equipment important to safety of a different type than previously analyzed in the USAR. With a fault tolerant system design that uses field proven components and software (that eliminated many existing single points of vulnerability thereby improving overall system reliability), the DEHC system will NOT increase the frequency of accidents previously evaluated in the USAR and will NOT increase the likelihood of a malfunction of equipment important to safety. There are no new system interfaces (PLU excluded) created by the proposed control system upgrade and no physical changes to the steam path, turbine-generator or steam bypass system. The new HMI interface that replaces hard controls perform the same control and indicating function and as such do not constitute a new interface. The design does not alter or affect any ECCS system or barrier credited in mitigating the consequences of an accident. As such, the proposed activity does NOT increase the consequences of an accident or malfunction of equipment important to safety as previously analyzed in the USAR and will NOT result in a design basis limit for a fission product barrier being altered or exceeded.

This 50.59 Evaluation has determined that the equipment/hardware changes, operating/maintenance procedure changes and modification/operability testing being implemented in conjunction with the proposed activity do not affect or alter the performance requirements or design function of the turbine and steam bypass pressure control system, Reactor Protection System, Turbine Generator, or any other SSC as described in the USAR. These changes have no adverse effect on how any USAR described design function is performed or controlled, and no adverse impact on plant procedures or system operating parameters. There are no changes to any USAR described evaluation methodology or the use of an alternative methodology in establishing the design basis or safety analyses. The turbine trip function is unchanged as is the reactor recirc pump EOC-RPT trip function. While the PLU function still actuates due to the same conditions (load reject) as originally designed, this function will now trip the turbine. The trip will fast close the control valves and intercept valves prior to stop valves and intermediate stop valve closure and thus presents itself to the reactor (pressure) in the same way as a fast closure from PLU actuation. Overspeed failure probability was provided by GE in the old design (Reference RBC-45208, Turbine S/N 170X662, Missile Analysis). Given the new evaluation produced a probabilistic failure rate for the new design, the method of evaluation is unchanged in that both are probabilistic and result in a similar conclusion of low failure rate; Therefore, the failure rate remains bounded by value stipulated in RBC-45208 as described in question 2. The proposed activity does not involve a test or experiment that would operate any SSC outside of its USAR described design function. A change to the USAR description of the overspeed trip system is required based on conversion of the mechanical overspeed to electrical-but-diverse overspeed trip however no changes to the Technical Specifications or Operating License are required. LBDCR 10.02-020 incorporates the changes to the USAR. Based on the results of this review, the proposed activity can be implemented without prior NRC review and approval.

Is the validity of this Evaluation dependent on any other change? ☐ **Yes** ☒ **No**

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ **Yes** ☒ **No**

Preparer: David Anders / See AS / DP Engineering / Design / 4-18-16
Name (print) / Signature / Company / Department / Date

Reviewer: Barry Burmeister / See AS / EOI / Licensing / 4-19-2016
Name (print) / Signature / Company / Department / Date

OSRC: Joseph Clark/See AS/4-28-2016
Chairman's Name (print) / Signature / Date

RBS 2016-006
OSRC Meeting #

II. 50.59 EVALUATION

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR?

☐ Yes
☒ No

BASIS: The turbine and steam bypass pressure control system is necessary for plant operation but does not perform any safety-related or safe shutdown function. The system is considered "Important to Safety" as it controls reactor pressure by regulating main steam pressure through modulation of the turbine control or steam bypass valves. Failure of the system does not compromise any safety-related system or component or prevent a safe shutdown of the plant. Therefore, the safety related equipment and functions described in the USAR are not affected.

Malfunctions of turbine and steam bypass pressure control system are the initiators of plant transients that are analyzed in USAR Sections 15.1.2, 15.1.3, and 15.2. These plant transients are as follows:

- Feedwater Controller Failure – Maximum Demand
- Pressure Regulator Failure — Open (Increase in Steam Flow)
  (Note: The pressure regulator failure-open is no longer a postulated single failure scenario; see LBDCR#10.02-020)
- Pressure Regulator Failure — Closed (Increase in Reactor Pressure)
- Turbine Trip with Bypass System Operation
- Turbine Trip with Bypass System Failure
- Generator Load Rejection with Bypass System Failure
- Generator Load Rejection with Bypass System Operation

These events are defined as Anticipated Operational Transients (events of moderate frequency) in USAR Table 15.0-1 and involve either an increase in steam flow to the turbine or increase in reactor pressure that may result from the failure of the turbine and steam bypass pressure control system.

Valve Response Times

The new DEHC system will not change the function or response parameters regarding these transients. The impact of the DEHC system upon these transients is a function of the Turbine Control and Bypass Valve response. Based on results from a nearly identical Ovation® TCS delivered to the Limerick Generating Station, the new DEHC bypass valve fast opening timing responses are within the design requirements and transient analysis (Reference: LGS Unit 1 and 2 Turbine Control System Upgrade, Bypass Valve Opening Delay on Turbine Trip, WNA-AR-00305-GLIM). Actual Limerick plant data supporting the bypass valve response time requirements are documented on pages 217 and 218 of the Modification Acceptance Test (MAT 12-00019-1). Additionally, the turbine control and valve characteristics including bypass control and valve characteristics used in cycle specific reload analyses were evaluated for potential impact to transient analysis and concluded that no changes are necessary due to the turbine control system upgrade. This analysis while supported by industry data is being verified by validation of RBS plant specific steam valve response time testing that will be performed under the Post Modification Test Plan (PMTP) for EC-54958.

Analog to Digital Conversion

The transition from analog to digital control introduces new failure modes; however, the failure affects are the same as previously analyzed in the USAR and the frequency of these events does not increase. Westinghouse performed a Failure Modes and Effects Analysis as documented in W120-0312 and Software Hazard Analysis 7231.015-001-001 that analyzed the new hardware and software failure modes. NEI 01-01 was utilized in development and analysis of the digital upgrade documented under the Software Hazards Analysis 7231.015-001-001. It was determined that that single failures and common cause failures of the new system would not result in plant transients that are not bounded by the USAR accident and transient analyses. The new DEHC system is a more reliable system due to its component

redundancy, online error checking, fault diagnostics, and does not increase the frequency of occurrence for a system malfunction. Protective actions such as reactor scram required to respond to an EHC failure are provided by other systems external to EHC and the EHC control interfaces and remain unaffected by the change. The proposed change does not alter or affect the Reactor Protection inputs for Turbine Control Valve Fast Closure or Turbine Stop Valve Closure and does not affect the reactor recirc pump (EOC-RPT) function.

The existing Electro-Hydraulic Control (EHC) system consists of two (2) analog loops for pressure control, a primary and backup pressure regulator. The design is redundant but is susceptible to single failures that have resulted in reactor SCRAMs due to inadvertent steam valve operation. .

In comparison, the new Ovation® Redundant Controllers are configured with redundant power supplies and dual-attached network interfaces. Each functional processor in the redundant pair executes the same application program, although only one processor at a time accesses the I/O and runs in the control mode. The secondary (partner) processor runs in either the backup, configure, or offline mode. The controllers are designed for "bumpless" automatic failover to the back-up controller when a fault is detected. Application software, point database, and configuration information is retained in non-volatile flash memory on the processor module.

The Ovation® system performs self-tests and online diagnostics that are capable of detecting various failure conditions down to the individual component level. For example, the system is capable of identifying failures of I/O cards, buses, power supplies, functional processors, and network communication issues. These features identify the presence of a fault, and determine the location of the failure to a replaceable module level. Therefore, for any component that is redundant, the design assures that a single active component failure within the new DEHC system will not result in a loss of control signals to Turbine Valves, Bypass Valves or MCR HMI displays.

The redundant control elements in the new system are configured to allow unrestricted operation with a single failure, and to facilitate on-line replacement of a failed component. For those components that are not redundant in design or are inputs from other systems, the new DEHC system provides a great deal of diagnostics and failure recognition, enhancing failure detection and thus reducing the probability of spurious steam valve motion or a turbine trip occurring. In the event one of these components was to fail outright or the signal was lost, the system is designed to fail safe upon loss of signal. In comparison, the existing analog EHC system was not-fault tolerant and did not provide the fault detection and failure response capability of the new DEHC system and therefore was more susceptible to inadvertent steam valve motion or turbine trips that would result in reactor SCRAMs.

Environmental Considerations

The new DEHC components are rated for the environmental conditions. (e.g. temperature, humidity, pressure, and radiation) for their intended locations such that they will operate within their specified tolerances without any degradation in performance over their expected life. The modification of existing cabinets, the installation of new cabinets, mounting of HMI displays in the Main Control Room, and the routing of new raceways and conduits are designed to meet Seismic Category II/I requirements where applicable.

Electromagnetic Compatibility (EMC) testing (i.e., emissions and susceptibility) was performed as documented in W120-0345 "Ovation® Platform Equipment EMC Qualification Summary Report" to demonstrate compliance of the Westinghouse Ovation® Platform equipment to EMC testing requirements of Regulatory Guide 1.180. Some new components installed were not tested per the EMC qualification summary report W120-0345; However, these components satisfactorily meet CE mark testing standards and/or other accepted industry EMC testing standards. Susceptibility values contained in report W120-0345 were compared against surveys of the areas (reference 6231.015-001-011) where equipment is to be installed and found to be acceptable both for radiated emissions to the environment and susceptibility from existing environmental EMI/RFI.

Based on the readings obtained at River Bend, the AMS concludes that the electromagnetic environment of the areas were components of the Ovation system are to be installed is comparable to that of other

nuclear power generation facilities and is bounded by the levels used to generate the immunity test levels in the EPRI and NRC EMC guidance documents. None of the emissions tests exceeded the EPRI TR-102323 Allowable Plant Limit, which is the critical level of concern with regards to equipment which is to be installed in a nuclear power plant environment.

Testing performed by Westinghouse and documented in W120-0345, "Ovation Platform Equipment EMC Qualification Summary Report" was done to demonstrate performance of the Westinghouse Ovation Platform equipment to following EMC requirements:

- Radiated and conducted Emissions limits in accordance with the guidelines of Regulatory Guide 1.180 Revision 1
- Radiated and conducted Immunity limits in accordance with the guidelines of Regulatory Guide 1.180 Revision 1

Also, Regulatory Guide 1.180 Revision 1 requirements are supplemented by EPRI TR-102323, Revision 3 where appropriate.

In short, the following sections provide details for, the I/O modules, power supplies, line filters used in H13-P821 and TMB-PNL01 to show they meet or exceed the qualifications required for the environments based on the test report's "EMI RFI Report from AMS at RBS" conclusion that none of the emissions captured at river bend exceeded the EPRRI TR-102323 Rev 1 allowable plant levels. That coupled with the implementation of the recommended Ovation grounding will ensure that the equipment will continue to operate under the normal plant conditions and not impact surrounding equipment. It is further noted that the use of hand-held portable radios is administratively prohibited in the Control Room and at TMB-PNL01. All equipment susceptible to RFI is installed in locations that either prohibit the use of hand-held radios or are sufficiently remote or protected from expected radio communications. In accordance with good design practice, all added instrumentation cables contain twisted pairs with individual pair shields that are connected to the plant shield ground bus.

RG 1.180 does not have a category for equipment important to power production. It does not address non safety-related equipment; However, the Regulatory Guide endorses the same testing for non-safety-related equipment that can affect safety-related equipment per the following excerpt: "While non-safety-related systems are not part of the regulatory guidance being developed, control of EMI/RFI from these systems is necessary to ensure that safety-related I&C systems can continue to perform properly in the nuclear power plant environment. When feasible, the emissions from non-safety-related systems should be held to the same levels as safety-related systems." MIL-STD-461D addressed test methods for EMI testing of electronic equipment. MIL- ST-461E merged documents MIL-STD-461D and MIL-STD-462D and superseded them. Therefore, the testing criteria and methods of MIL-STD -461E are the same as those in MIL-STD 462D.

The Ovation Platform Equipment EMC Qualification Summary Report documents that the EUT (equipment under test) was tested to the required Emission and Susceptibility tests. The test criteria for EUT is that it shall operate continuously during the EMC susceptibility tests with a Level "A" performance criteria as defined in EN 50082-2, which is defined as follows: The apparatus (equipment) shall continue to operate during and after the test. No degradation or loss of function is allowed below a performance level specified by the manufacturer when the apparatus (equipment) is used as intended. This is a higher level of performance than required for non-safety related turbine equipment that does not need to continue to operate during and after an event.

Therefore, the new digital DEHC system will have higher reliability than the previous analog EHC system due to redundancy, fault tolerance, and improved maintainability. The improvement in system reliability will not increase the frequency of occurrence of the accidents postulated in the USAR sections noted above.

Hard Control to Soft Control Conversion

The conversion from hard controls to soft controls involves more than minimal difference in the HMI;

however, the new controls will not introduce any new failure affects as previously analyzed in the USAR and the frequency of these events does not increase. The TCS will use a graphical operator interface and soft control graphics, which are designed to be intuitive and straight forward for the plant operator to navigate. These graphics are accessed via the Operator Work Station (OWS). The HMI interface includes graphic displays with monitoring and control functions. The functions are provided with additional system diagnostics in a manner that does not reduce the ability of the operator to operate the system or respond to system transients.

The Ovation based Digital Control System (DCS) infrastructure consists of networking equipment and the Ovation network, along with engineering servers, historical data storage and maintenance and engineering workstations, report servers, database and system servers, and data link servers. Each component in the infrastructure performs a specific service within the DCS. The windows into the Ovation system are the Ovation Workstations. These are high performance Windows based machines that allow for full-multitasking operation and a Common Desktop environment. The system is configured with varying levels of security protocol / access to the DCS with general functionality as follows:

- The Ovation Operator Workstations (OWS) provide the operator control and monitoring functions. The Ovation Operator software provides a dynamic view of all plant processes which include control graphics, diagnostics, trends, alarms, and status displays. Access to dynamic system points, historical data, general messages, standard function displays, event logging, and alarm management programs are available through operator navigation tools. Integrated into the Ovation process (graphic) displays are the soft control functions which work in parallel with the hard-wired plant controls. The Ovation Operator Workstation is supplied with a touchscreen, keyboard, mouse and trackball. The OWSs are located in H13*P680 and the UO/ATC station.
- The Ovation Maintenance and Engineering Work Station (EWS) contain the Ovation System Database. This database maintains the software and configuration data for all the workstations and controllers. The Ovation System contains one master database and multiple distributed databases. When data is modified in the master database, changes are broadcast over the network to the distributed databases located on every drop. The EWS are located in the new infrastructure panel (C91-P609) and the existing ERIS infrastructure cabinet A, via the use of virtual machine architecture.

The majority of the Main Control Room hardwired alarms were maintained with two additional hardwired alarms added to H13-P680 for Major and Minor Trouble. The two additional alarms serve as common alarms for DCS generated alarm points and are used to alert Operations to system issues and severity. The DCS system HMI screens will detail specific alarm points initiating the common alarms. One alarm eliminated was the Turbine Trip Bypass alarm. The Turbine Trip Bypass and associated alarm was installed several years ago under MR89-0046 as part of a workaround modification to bypass turbine trips during turbine testing evolutions and was designed to eliminate nuisance tripping experienced at RBS during test evolutions. The bypass trip switch and alarm are no longer required due to the new testable dump manifold configuration and controls installed to eliminate nuisance trips associated with testing and provide a more stable testing platform from which to conduct tests. The remainder of the hardwired alarms are unchanged with some names changing slightly to reflect the new control system. The alarm functions, actuations and logic are identified in vendor manual W120-0311 Section 3.39 and Section 5.23. The Input/Output database, W120-0319, identifies the alarm points associated with external and internal alarms that are indication on the Ovation HMI alarm screens.

The HMI displays provide several advantages over the previous "hard-wired" controls. The HMI allows the Operator to view multiple Process Display Graphics for different functions (e.g., Turbine Valve testing). In accordance with Human Factor procedures, background color changes provide the operator with a system status that is easily visible throughout the control room. The interface provides a system status graphic that displays the status of all drops connected to the process control network. This system status graphic indicates by successively focusing on the location and nature of a malfunction, down to smallest replaceable component. This status graphic is also capable of indicating the communication status of any data link. Graphic symbols for the drops change color to indicate malfunctions.

All of the existing turbine and pressure control functions will be available on both of the operator

workstations and provide the operator with the same controls for startup, power generation, testing and shutdown activities as the existing system. The HMIs will use the same graphics (i.e., exact same screens). All critical functions require a confirmation step prior to execution of the command. The software fields are constrained, such that the system does not allow the insertion of values (e.g., setpoints, rates, etc.) outside of the minimum/maximum limits and will display an INVALID message if attempted by the operator. This software constraint feature ensures that Operators do not enter (for example) an automatic cool-down rate greater than the applicable Tech Specs. The graphic displays and control features of the HMI workstations have been developed in accordance with industry principles outlined in NUREG 0700 "Human-System Interface Design Review Guidelines," U.S. Nuclear Regulatory Commission, May 2002. The graphics are intuitive, straightforward to use, and easy to navigate between system displays and meet the human factors requirements of NUREG 0700.

The existing EHC system controls (switches, pushbuttons and indicators and EHC system panel inserts) on the Main Control Room Turbine Console and Turbine Vertical Board (H13-P680 inserts 7 and 8) in addition to the Turbine Front Standard Test Panel (H13-P870 insert 54D) will be replaced with functionally redundant and independent HMI Display Monitors and redundant, manual turbine trip pushbuttons located at H13-P680 Inserts 7 and 8 respectively. The two workstations may have different displays that may be viewed for quicker operator understanding of the turbine conditions. The operator workstations will communicate with the Turbine and Pressure Controllers over redundant data highways. Since the TCS / ETS program is running on the controllers located in H13-P821 / TMB-PNL01 and not the HMI displays, failure of one HMI display monitor, both HMI display monitors, or the data highway network will have no effect on turbine and pressure regulator control as the controllers will still have the ability to operate independently. The redundant data networks and HMI displays are powered by separate UPS systems so that station blackouts and loss of offsite power will not result in the immediate interruption of HMI indication and control function. Also, a single power source failure will not result in loss of the HMI display indication and control features. The hardwired interface to the Emergency Response Information System (ERIS) will be maintained, which would allow the operators to monitor bypass, control valve, and main stop valve positions and 2 of the 3 throttle pressure transmitter pressures.

At both the turbine front standard locally and remotely in the MCR at H13-P680, two (2) new manual Turbine Trip pushbuttons are provided and configured in a two-out-of-two logic scheme using a mechanical push-pull arrangement to prevent inadvertent actuation. Operating an individual Turbine Trip pushbutton provides feedback (via a spare contact that is wired to the DEHC and alarmed) that the associated pushbutton is actuated (pushed in). The Turbine Trip pushbuttons are located next to each other and are depressed simultaneous to manually trip the Turbine. These pushbuttons are hardwired in series with the Emergency Trip System (ETS) outputs, which trips the turbine by de-energizing the solenoids in the Testable Dump Manifolds. These pushbuttons provide the operator with the ability to trip the turbine even in the highly unlikely event that one or both workstations or data highway networks fail. Note that in the original MK II front standard design, a single handle was used was used to trip the turbine locally at the front standard. In the new design the same two-out-of-two pushbuttons are used, but fitted with a protective cover to prevent inadvertent actuation, providing Operations with the same ability to trip the turbine locally from the Turbine Front Standard.

In summary, the new Operator interfaces do not result in more than a minimal increase in a frequency of occurrence of an accident previously evaluated in the USAR based on capabilities and the redundancy of the OWS, along with the hardwired, manual turbine trip pushbutton logic scheme. The turbine overspeed protection system instrumentation and turbine control valves are designed to stop admitting steam into a turbine when the turbine control system has failed to maintain speed control to protect the turbine from excessive speed. While nuclear plants do not classify this system as "safety-related," the system plays an important role in the plant, protecting both personnel and reactor systems from the missiles that can result when turbines disintegrate at higher than normal rotational speeds. The turbine overspeed trip should prevent loss of turbine speed control from creating significant overspeed events.

Overspeed Protection

General Design Criterion (GDC) 4 requires that structures, systems, and components (SSCs) important to safety be protected against the effects of missiles that might result from equipment failures. Therefore, protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment, or structures. As documented in RBC-45208, GE states that under a total loss of speed control, the turbine is incapable of reaching speeds required to result in missile generation. In addition, RBS has a favorable orientation of the turbine meaning its shaft is perpendicular to safety related structures, systems, and components (SSCs) reducing the likelihood that should missile generation occur, safety related SSCs would be damaged. The existing EHC System provides the following three (3) barriers to prevent turbine overspeed:

- Normal EHC Speed Regulation Controls
- Mechanical Overspeed Trip (throw out bolt)
- Electrical Backup Overspeed Trip

The new DEHC System will provide the following three barriers to prevent turbine overspeed:

- Normal DEHC Speed Regulation Controls (Ovation®)
- Primary Overspeed Trip initiated from the DEHC System ETS by a Woodward® Diverse Overspeed Protection Subsystem (DOPS) using three passive speed probes
- Backup Overspeed Trip initiated from the DEHC System TCS by an Ovation® Speed Detection Subsystem using three active speed probes

The Woodward ProTech® DOPS is a dedicated electronic overspeed protection system which is independent and electrically diverse from the Ovation® TCS control and overspeed protection system. The DOPS has independent speed sensors, trip modules, power supplies and actuates an independent Testable Dump Manifold (via 125VDC solenoids) to trip the turbine. The overspeed trip setpoint for this primary overspeed trip will remain at 110% of rated turbine speed (setpoint unchanged from the previous mechanical overspeed trip).

The DEHC System Ovation® TCS System performs normal speed regulation and provides the backup overspeed trip. The Ovation® TCS Speed Detection Subsystem is independent and electrically diverse from the DOPS. The Ovation® TCS Speed Detection Subsystem has independent speed sensors, trip modules, power supplies and actuates an independent Testable Dump Manifold (via 24 VDC solenoids) to trip the turbine. The backup overspeed setpoint for the new system will remain at 111.5% of rated speed of the turbine (setpoint unchanged from the previous electrical backup overspeed trip). The TCS and ETS Testable Dump Manifolds, which include two-out-of-three hydraulic trip assemblies, are redundant and designed to ensure that no single component failure will prevent a turbine trip.

For additional information of overspeed protection, refer to question 2 response on overpseed protection of this evaluation.

Conclusion

As such, the replacement of the functionally diverse mechanical overspeed trip and electrical backup overspeed trip with redundant electrically diverse trips will not increase the frequency of the occurrence of an accident previously evaluated in the USAR. Therefore, the proposed activity, which includes the transition from an analog to digital control system, the transition from hard to soft controls and the replacement of the overspeed trip systems will not increase the frequency of occurrence of any of the Anticipated Operational Transient evaluated in the USAR or the probability of turbine missiles resulting from a turbine overspeed event.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR? ☐ Yes ☒ No

BASIS: The EHC control system being replaced is a non-safety related system, but has support functions that are important to safety. The new DEHC is designed to be more reliable than the existing system by using a field-proven architecture and hardware/software components, by eliminating single points of failure through the addition of redundancy where practicable, and by performing extensive vendor and operational testing. The new system retains all of the control functions of the existing system. There are no new control functions that directly interface with important to safety SSCs. Reactor scram functions based upon turbine stop valve closure and low ETS fluid pressure (i.e. turbine control valve fast closure) are not affected by this modification. A direct power load unbalance (PLU) turbine trip is being added, however this is not an adverse change as discussed in the PAD and maintains the same effect (load reject) on reactor pressure and temperature as previously evaluated in the USAR. The function of the replacement system will not change the operating or design parameters of any plant system.

Seismic Requirements

The installation of the new digital equipment occurs in areas that contain important to safety SSCs, and the appropriate design features have been included to preclude the digital system from causing the malfunction of surrounding components. The modification of existing cabinets, the installation of new cabinets, mounting of HMI displays and keyboards in the Main Control Room, and the routing of new raceways and conduits have been evaluated to ensure the design meets Seismic Category II/I requirements, as applicable. These components are classified as Seismic II/I component; that is, the DEHC equipment should not become unstable or undergo gross collapse under the effects of the postulated seismic responses due to the Safe Shutdown Earthquake (SSE). As noted below, EMC emissions have been tested to ensure they are in compliance with station requirements.

Effect on Power Supplies and HVAC

There is adequate capacity of non-Class 1E power systems to accommodate the DEHC component electrical loading without significant reduction in margin or potential adverse impact to other station loads. The impact of thermal loading of DEHC components on the existing HVAC system has been evaluated to be well within their design limits with insignificant impact on thermal margins. The only exception is that of the ERIS room temperature during a loss of offsite power (LOOP) event or Station Black Out (SBO). During a LOOP or SBO, the DEHC components installed in this room are not required to operate. In order to address this existing deviation, the DEHC infrastructure equipment in the ERIS room may need to be manually shutdown in order to maintain room temperatures within thermal margins. Refer to CR-RBS-2016-02737 for disposition and potential actions taken to address this issue.

System Redundancy

The DEHC system architecture utilizes redundant fiber optic cables housed in dedicated conduits between the Emergency Trip System (ETS) and Turbine Control System (TCS) redundant controllers installed in EHC Cabinet 1H13-P821 and the TMB-PNL01 Remote input/output (I/O) TCS and ETS cabinet located on the turbine deck. Fiber optic cables are not susceptible to electromagnetic or radio frequency interference (EMI/RFI). Therefore, a single failure of the data communications link or a strong source of EMI/RFI between these portions of the control system will not cause a disturbance within the TCS or ETS. The controllers and remote I/O modules are configured with two redundant power supplies. This redundancy scheme is extended to the two independent AC power sources (inverters) feeding cabinet power supplies. Communications switches, Ethernet cables, and Fiber used throughout the network are redundant as well. This configuration arrangement ensures that no single electrical failure causes a loss of any redundant equipment.

Redundant Power Supplies
The DEHC is a non-safety related system and is powered from non-Class IE power. This modification implements a modified power scheme that maintains system redundancy as exists in the present design. However those portions of the system previously powered by the 120VAC permanent magnet generator (PMG) will now be supplied by the existing redundant inverters (battery backed uninterruptible power) used to power the remaining portions of the current EHC equipment. All power, control, and signal cabling is routed in non-Class IE raceway or conduit. Power and control cables are separated from low voltage signal cabling. Grounding details for electronics comply with station and industry standards to the extent practical to maintain conformance to EMC standards for susceptibility and emission.

Environmental Considerations
The cabinets/panels have been evaluated against the specific environmental parameters (i.e., temperature ranges) for their respective locations and determined to have adequate cooling to support the DEHC components well within their thermal limits (with the exception of ERIS room temperatures during a LOOP as described above). The DEHC electronics are maintained within their specified environmental envelope to ensure component reliability and component life. The inherent reliability of the DEHC system architecture (incorporating redundant microprocessor controllers, I/O and communications buses powered by redundant power supplies) all decrease the probability of a malfunction of the DEHC system caused by hardware failure associated with any single component. The high reliability and low failure frequency of the DEHC is validated by the fault tree analysis performed to support the failure mode and effects analysis (FMEA) that was completed for the new DEHC (Reference W120-0312). The FMEA determined that the DEHC modification does not introduce any new failure modes that have been evaluated in the USAR. The FMEA, in conjunction with the 1) Software Hazards Analysis (Reference 7231.015-001-001), 2) the DEHC Reliability Analysis, and 3) the Overspeed Protection Fault Tree Analysis (Reference 7231.015-001-001), provide an overall conclusion that the DEHC modification does not significantly alter the probability or consequences of events analyzed in the USAR.

Diagnostics and Error Checking
In addition, extensive online diagnostic and error checking alert the operators to hardware failures and data communication errors via Operator Workstations in the MCR and the Maintenance & Engineer Workstation located in the Infrastructure Cabinet ensuring that component failures are identified and promptly addressed.

Software Common Cause Failure
Digital control systems are susceptible to postulated Software Common-Cause Failures (SWCCF): however, the failure affect is the same as the EHC System. Per the Westinghouse software hazards analysis of the Ovation® platform (document 7231.015-001-001), the failure mode of a postulated SWCCF for the DEHC system is the same as the failure effects of the analog system. The existing analog system is limited in its redundancy of components compared to the new DEHC system. With the more reliable DEHC System, the likelihood of a malfunction of equipment important to safety does not increase. The DEHC microprocessor based system is programmed to perform all logic and computational operations for the turbine and steam bypass pressure control system, including all monitoring and equipment control. Westinghouse/Emerson has created application software for control and protection algorithms for data acquisition, control sequencing, and displays for each application. The Ovation® software has been used for numerous nuclear and industrial applications and proven throughout the steam turbine industry. A detailed Validation and Verification process has been performed, as part of this design project, which is consistent with microprocessor technology industry standard requirements, and includes software in loop testing. In addition, factory acceptance testing, on-site acceptance testing, and post modification testing to assure software integrity and system performance will be performed prior to placing the system in service as verified by the Post Modification Test Plan (PMTP) EC-54958.

The likelihood of a common-cause failure (SWCCF) is considered extremely small based on compliance to industry standards, regulations, design processes, and qualification of platforms (hardware and software). The Ovation® platform and DEHC specific applications have been designed, tested, and controlled using established software development practices and methods. Quality Controls and other factors such as Software Development Lifecycle and Process, Software Architecture and Version Control,

Supplier Performance Records, Software Validation, Software Test Plans, and Software Configuration Management plans ensure that the software delivered and installed can be reasonably expected to perform dependably.

Commercial Off the Shelf Components (COTS)

The Ovation® system minimizes failures by using industry standard components that are supplied by vendors with proven reputations for providing high quality products. The Westinghouse operating experience base provides evidence that the Ovation® based Distributed Control System (DCS) platform is highly reliable, fault tolerant, and in the worst-case scenario, it is designed to fail to a safe state.

The commercial-off-the-shelf (COTS) microprocessor and operating system core technologies used in the Ovation® system were proven within other industries before they were introduced to Ovation®. The Ovation® platform technologies, which are the generic hardware and software components developed to form the real-time control system, were field proven in non-nuclear industries prior to being implemented for nuclear plants. Ovation® system user groups share operating experience and Westinghouse provides notifications of hardware/software problems via technical information notices. Because of this information sharing between users and the vendor, the chances that a common cause failure could exist and remain undetected is greatly reduced. Any new failure mechanism is bounded by the system failures currently analyzed and will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR.

Digital Asset Classification

The DEHC is a Critical Digital Asset (**CDA**) and is classified as a Level 4. The Architecture is such that data may only pass from the DEHC System (Level 4) to another Level 4 Data Acquisition Network (i.e., Emergency Response Information System (ERIS) Computer). Physical security is achieved as the system equipment is mounted within cabinets that are secure during normal operation of the facility. Administrative procedures/processes are in place to limit download of software to/from the platform and limit access to the software via security passwords. A separate engineering change, EC-63931, will initiate any Cyber Security changes required as determined by the review.

Emissions and Susceptibility Testing

Digital and analog equipment utilized for I&C systems are typically sensitive to EMI and RFI. Therefore, the I&C system equipment, specifically digital system hardware, is required to demonstrate immunity from EMI/RFI to ensure required functionality and accuracy during normal plant operation. In addition, the conducted and radiated emissions from the equipment must be within industry standard limits in order to not impact other EMI/IRFI sensitive equipment installed in the vicinity.

Open frame testing performed by Westinghouse and documented in W120-0345 "Ovation® Platform Equipment EMC Qualification Summary Report" demonstrates the performance of the Westinghouse Ovation® Platform equipment to satisfy technical requirements provided in US NRC Regulatory Guide 1.180:

- Radiated and conducted Emissions
- Radiated and conducted Immunity

(Note: Regulatory Guide 1.180 requirements are supplemented by EPRI TR-102323, where appropriate.)

The open frame configuration exposed the equipment to the most adverse test conditions. The EMC Type Tests completed demonstrate Ovation® platform equipment compliance to the various EMC requirements. Other system components (e.g., HMI Monitors, Media Convertors, Workstations, Root Switches, etc.) were tested to applicable EMC standards as documented in the Engineering Change. Therefore, the proposed activity does not result in an increase in the likelihood of occurrence of a malfunction due to EMI/RFI effects of an SSC important to safety previously evaluated in the USAR.

Overspeed Protection

The existing turbine mechanical overspeed system is being replaced with a diverse, digital turbine overspeed protection system (DOPS). The new DEHC system contains two digital overspeed trip systems that are independent and electrically diverse such that the primary110% overspeed trip and

backup 111.5% overspeed trips are not susceptible to a SWCCF that would render them both inoperable. Diversity between the Woodward ProTech® primary overspeed trip and the Westinghouse Ovation® backup trips (via the speed detector modules in the TCS remote I/O cabinets and controller application software in the ETS) is achieved by the following:

- Human Diversity; different companies, therefore different design, application, certification, and test organizations
- Design Diversity; different architectures
- Software/Firmware Diversity; different programming and configuration tools
- Signal Diversity; Active versus Passive speed probes
- Equipment Diversity; different manufactures of fundamentally different designs

The Woodward ProTech® is certified as an IEC-61508 Safety Integrity Level 3 (SIL-3) system, provides 500VAC isolation (from passive probe input to chassis and to all other circuits) and meets the following EMC standards:

- Emissions: EN61000-6-4
- Immunity: EN61000-6-2

Testing of the electronic trips can be performed by lowering the overspeed setpoints. Therefore, by eliminating the mechanical trip, the turbine trip system does not need to be tested at speeds above the operational range of the turbine.

The upgraded system performs the same basic functions of the existing analog EHC system, using the same process inputs (e.g., throttle pressure, turbine speed, generator current, intermediate pressure, discrete inputs). The upgraded system has eliminated the turbine overspeed mechanical trip function, and replaced this function with a diverse electronic overspeed trip system with its own dedicated testable trip manifold (TDM), with the same overspeed trip setpoint of approximately 110%. The backup electrical overspeed trip system trips the turbine at approximately 111.5% overspeed through its associated TDM. In addition, each of these overspeed trip systems provides cross trips to each other. The effect of the upgraded system on the probability of failure to trip was calculated using fault trees as documented in 7231.015-001-001. The elements of the analytical methods that are described in Chapter 10 Steam and Power Conversion System of the USAR are customer proprietary information not available to this Evaluation but are probabilistic in nature and the current missile probability analysis for the low pressure turbine monoblock rotors discussed in RBC-45208 remains bounding.

Westinghouse report 7231.015-001-001 determined that the new DEHC system (including the Woodward ProTech® DOPS) has a lower probability of complete control system failure on demand (PFD) than the current analog system. Report 7231.015-001-001 uses a probabilistic methodology acknowledged by the NRC in NUREG-0492 for fault tree analysis. As stated in chapter 3.5 of the USAR an annual probability of complete system failure is 10-8. The new overspeed reliability analysis report 7231.015-001-001 documents a 5.41 x 10-9 annual probability of complete failure which is less probable than the current rate as described in the USAR. Therefore, the proposed change does not result in an increase in the likelihood of occurrence of a malfunction of an SSC important to safety described in the USAR.

Conclusion

Therefore, the proposed activity, which includes the transition from an analog to digital control system, the transition from hard controls to HMI graphics and the replacement of the overspeed trip systems will not result in an increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR? ☐ Yes ☒ No

BASIS: The EHC control system being modified by this change is a non-safety related system that has no credited function for accident mitigation. The proposed activity does not establish any new functions, interfaces or design parameters with components or systems that are credited for accident mitigation or barrier performance. The DEHC system may be the initiator of a number of plant transients that are analyzed in the USAR, but the proposed activity does not alter any components or systems credited in the accident analyses. As such, a failure of the DEHC system will not result in an increase in offsite dose or the dose to operators during accident response and the proposed activity will not result in an increase in the consequences of any currently analyzed accident in the USAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR? ☐ Yes ☒ No

BASIS: The proposed activity involves the replacement of the EHC system with a more reliable DEHC system. There are no new USAR described design functions or system interfaces that result from this upgrade. There is no increase in the probability, type or effect of a malfunction of equipment important to safety. The more reliable digital system will improve system performance, reduce transients in the turbine controls and reduce challenges to systems important to safety. Failures associated with the DEHC system that impact the reactor are those that either cause an increase or decrease in steam flow to the turbine or bypass valves. The bounding conditions for these events are currently analyzed in the USAR and the consequences resulting from these transients, as currently analyzed, are not affected.

5. Create a possibility for an accident of a different type than any previously evaluated in the USAR? ☐ Yes ☒ No

BASIS: The new DEHC does not create an environment which may adversely affect other equipment. The DEHC installation is consistent with Seismic and EMI/RFI design requirements and will not affect other equipment. The electrical and HVAC impact of this upgrade (with the exception of ERIS room temperatures during a LOOP / SBO as described above) is within the capability of the current systems. The DEHC system performs the same function as the EHC system as described in the USAR but in a more reliable manner. The proposed activity does not create any new functional interfaces nor does it change the operating parameters of any system as described in the USAR. There is no new interface between this system and systems providing reactivity control, pressure boundary maintenance (isolation), residual heat removal or emergency cooling water supply to the Reactor Pressure Vessel.

The basis for the most limiting USAR failure mode as it relates to turbine controls remains unchanged. USAR Chapter 15.0 lists the transients that could produce a significant reduction of the minimum critical power ratio (MCPR). The first transient listed is "Turbine Trip without Bypass or Generator Load Rejection. Another transient listed is "Feedwater Controller Failure". Note for feedwater controller failure the Tech Spec 3.7.5 basis states that the feedwater controller failure to maximum demand (USAR 15.1.2) is an event for which BPVs are expected to respond to preclude problems with MCPR, APLHGR, & LHGR. BPVs will continue to be able to satisfy the TS/TRM requirements of initiating opening of the BPVs within 0.100 seconds and that the BPVs are at >80% capacity in less than 0.300 seconds. Ref: TS SR 3.7.5.3, TRM 3.7.5.3, & STP-509-4801. USAR Section 15.1.2, 15.2.2, Table 15.0-1 and Table 15.0-1A were reviewed. It has been determined that the limiting transient is Main Generator Trip (Load Rejection) with Bypass System Failure. The MCPR for this limiting scenario is 0.11 and the basis is unchanged. Under the most limiting scenario (Main Generator Trip with Bypass System Failure), a Power to Load Unbalance (PLU) will be initiated that will actuate fast acting solenoids as previously occurred under the old system. This will close control valves and intercept valves by depressurizing the fast acting solenoid header. The fast acting solenoid header pressure reduction initiates an RPS reactor scram. Under the change, in addition to a PLU actuation, a new and simultaneous turbine trip is initiated however the turbine trip actions which result in all valves going closed is a slower function due to control loop execution time (PLU loop execution is 16ms vs Turbine Trip loop execution of 35ms), and stop valve stroke times (SVs are 100ms vs CVs at 80ms), therefore by the time SVs close, all steam flow through the turbine has ceased resulting in no change to reactor reactivity or pressure. This action also results in the same

reactor recirc pump (EOC-RPT) function as occurred prior to the change. The failure effects of the replacement DEHC system are bounded by the accident analysis demonstrated in Chapter 15 of the USAR.

Therefore, the proposed activity does not create the possibility for an accident of a different type than previously evaluated in the USAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the USAR? ☐ Yes ☒ No

BASIS: The DEHC system contains two separate sets of redundant digital controllers, with separate remote I/O inputs. The functions of reactor pressure control, turbine speed control, and turbine load control, and turbine backup overspeed trips are performed by redundant controllers in the Turbine Control System (TCS). The backup overspeed protection function in TCS uses active speed probes and speed detector modules (SDMs) to directly trip the Testable Dump Manifolds (TDM) independent of the digital controllers. The turbine trip functions are performed by a separate set of remote I/O inputs and redundant Emergency Trip System (ETS) controllers, and the primary overspeed protection uses separate and diverse passive speed probes and overspeed protection trip modules (i.e., DOPS). A postulated failure of both TCS controllers would result in a turbine trip with bypass failure. A failure of both ETS controllers will result in a turbine trip with bypass operation since bypass valve operation is performed by TCS. Furthermore, a hardwired manual turbine trip function is provided for the operator to trip the turbine.

The upgraded system has the same failure mode on loss of power as the system being replaced. Upon loss of power to the system, the turbine will trip and the bypass valves will not open. In response to a complete loss of power, the turbine would trip due to de-energization of the Testable Dump Manifold (TDM) solenoids. This would result in a turbine trip without bypass as is analyzed in Chapter 15 of the USAR. Upon power restoration:

a. The turbine will be in a tripped state
b. Speed setpoint will be tracking the speed feedback
c. Operations would need to set the pressure setpoint so bypass valves could control throttle pressure provided the condenser vacuum inhibit is not active as these setpoints will assume an offline tripped state.

Items b and c are different than the analog EHC system being replaced, where operator actions would be required. However, these are enhancements that would reduce operator work load. The consequences are bounded by the turbine trip with bypass failure event analyzed in Chapter 15 of the USAR.

The system responds to a reset condition (that is, loss of all power) in a fail-safe manner. All critical control outputs are configured to revert to a tripped state on initialization.

a. Modulating control outputs reset to the module low-scale state when the controller fails, and completely de-energize when power is lost.
b. Digital (on/off) outputs fail to their de-energized state either when the controller fails or when power is lost.
c. System control logic automatically maintains the turbine trip condition when controller power is restored and the system initializes so that the system behaves predictably in these instances.

The human-machine interface (HMI) was changed by introducing soft control graphic displays for operation. Failure modes are different from the existing system due to using digital technology, but the failure modes are bounded. The Manual Turbine trip interfaces continue to be hardwired but there is a soft manual trip button available on the HMI also. Failure of the soft HMI does not affect functionality of the hardwired controls. Failure of the soft HMI can result in turbine trip or failure to trip the turbine using the soft HMI control. Failure of the HMI resulting in a turbine trip are bounded by existing analysis. Failure of the HMI such that no trip occurs are eliminated using the hardwired controls available at H13-P680 or locally at the turbine front standard.

Thus, the failure effects of the new DEHC System are consistent with the failure effects of the existing EHC System and the results of these malfunctions are the same as previously evaluated in the USAR.

The function of maintaining adequate reactor pressure during a transient or accident remains unchanged through the use of the Main Steam Isolation Valves (MSIV) and Safety Relief Valves (SRV), which have no functional interface with the proposed activity.

Therefore, the proposed activity does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously analyzed in the USAR.

7. Result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered? ☐ Yes ☒ No

BASIS: The modification does not change the operating functions or parameters of the EHC system. The failure modes identified and evaluated for the system can either cause an increase or decrease in steam flow to the turbine. The bounding conditions for these events are not changed, since the valves, the fast acting solenoids, and the timing that controls the valves are not being changed. The modification does not impact the components credited for responding to these events. For all events analyzed, the fuel barrier integrity remains intact. This fission product barrier is not altered or exceeded.

The proposed upgrade does not interact directly with the components used to protect the reactor coolant pressure boundary and does not impact the design pressure of the boundary. This fission product boundary is not altered or exceeded. The modification does not involve the containment boundary or any containment isolation devices or signals. This modification does not alter or exceed the limit for the containment boundary. The operation of the replacement system does not impact maximum post accident pressure. As stated previously, USAR Chapter 15.0 lists the transients that could produce a significant reduction of the minimum critical power ratio (MCPR). The first transient listed is "Turbine Trip without Bypass or Generator Load Rejection. Another transient listed is "Feedwater Controller Failure". Note for feedwater controller failure the Tech Spec 3.7.5 basis states that the feedwater controller failure to maximum demand (USAR 15.1.2) is an event for which BPVs are expected to respond to preclude problems with MCPR, APLHGR, & LHGR. BPVs will continue to be able to satisfy the TS/TRM requirements of initiating opening of the BPVs within 0.100 seconds and that the BPVs are at >80% capacity in less than 0.300 seconds. Ref: TS SR 3.7.5.3, TRM 3.7.5.3, & STP-509-4801. USAR Section 15.1.2, 15.2.2, Table 15.0-1 and Table 15.0-1A were reviewed. It has been determined that the limiting transient is Main Generator Trip (Load Rejection) with Bypass System Failure. The MCPR for this limiting scenario is 0.11 and the basis is unchanged. Under the most limiting scenario (Main Generator Trip with Bypass System Failure), a Power to Load Unbalance (PLU) will be initiated that will actuate fast acting solenoids as previously occurred under the old system. This will close control valves and intercept valves by depressurizing the fast acting solenoid header. The fast acting solenoid header pressure reduction initiates an RPS reactor scram. Under the change, in addition to a PLU actuation, a new and simultaneous turbine trip is initiated however the turbine trip actions which result in all valves going closed is a slower function due to control loop execution time (PLU loop execution is 16ms vs Turbine Trip loop execution of 35ms), and stop valve stroke times (SVs are 100ms vs CVs at 80ms), therefore by the time SVs close, all steam flow through the turbine has ceased resulting in no change to reactor reactivity or pressure. This action also results in the same reactor recirc pump (EOC-RPT) function as occurred prior to the change. .

Therefore, the proposed activity does not result in a design basis limit for a fission product barrier being altered or exceeded.

8. Result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The upgraded system performs the same basic functions of the existing analog EHC system, using the same process inputs (e.g., throttle pressure, turbine speed, generator current, intermediate pressure, discrete inputs). Thus, the analytical methods described in USAR Chapter 15 Accident Analyses remain valid for the upgraded system, and is not affected by the upgraded system. The upgraded system has eliminated the turbine overspeed mechanical trip function, and replaced this function with a diverse electronic overspeed trip system with its own dedicated testable trip manifold (TDM), with the same overspeed trip setpoint of approximately 110%. The backup electrical overspeed trip system trips the turbine at approximately 111.5% overspeed through its associated TDM. In addition, each of these overspeed trip systems provides cross trips to each other. The effect of the upgraded system on the probability of failure to trip was calculated using fault trees as documented in 7231.015-001-001. The elements of the analytical methods that are described in Chapter 10 Steam and Power Conversion System of the USAR are customer proprietary information not available to this Evaluation but are probabilistic in nature and the current missile probability analysis for the low pressure turbine monoblock rotors discussed in RBC-45208 remains bounding.

As stated in chapter 3.5 of the USAR an annual probability of complete system failure is 10^{-8}. The new overspeed reliability analysis report 7231.015-001-001 documents a 5.41×10^{-9} annual probability of complete failure which is less probable than the current rate as described in the USAR. Therefore, the proposed change does not result in a departure from a method of evaluation described in the USAR used in establishing the design basis or in the safety analysis.

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

ATTACHMENT 1

SAFETY EVALUATIONS ASSOCIATED WITH

TEMPORARY CHANGES / PROCEDURES

SAFETY EVALUATION NUMBER: **None**

ATTACHMENT 1

SAFETY EVALUATIONS ASSOCIATED WITH

TECHNICAL REQUIREMENTS MANUAL
MAINTENANCE & LICENSEE AMENDMENT REQUESTS

SAFETY EVALUATION NUMBER: 2015-001 R1
2015-001 R2
2017-001

I. OVERVIEW / SIGNATURES[1]

Facility: RBS **Evaluation #2015-001 / Rev. #:** SE #2015-001 / Rev.1

Proposed Change / Document: EC62442

Description of Change: Engineering change EC62442 is the RBS Cycle 19 reload evaluation for the forced outage (FO-16-01). The core reload is a recurring activity for each fuel cycle. At the end of each fuel cycle, depleted fuel assemblies are discharged from the core and replaced by fresh reload assemblies. The remaining bundles resident in the core are shuffled to new locations and fresh fuel is loaded in accordance with the next cycle's core design and Reference Loading Pattern (RLP). This evaluation addresses the Cycle 19 reload changes for operation of the core including implementation of a Lead Use Assembly (LUA) program (GNF3 fuel), implementation of an Enhanced Lead Use Channel (LUC) program (NSF Channels), and for completeness the revision to the C19 RLP to replace a failed bundle. Note that the revision to the C19 RLP was performed under approved methods referenced in RBS TS 5.6.5; and the revision required no further evaluation beyond EN-LI-100 screening. As described in the preceding PAD (EC55796), this 50.59 evaluation was completed to document the impact of proposed changes to the RBS Core Operating Limits Report (COLR) which reflect C19 core operation as impacted by GNF3 LUA and NSF LUC.

EC55796 addresses changes associated with the Cycle 19 reload core into which 220 fresh GNF2 bundles and 4 fresh GNF3 LUA, fabricated by Global Nuclear Fuel (GNF), were loaded during RFO18. The GNF3 LUA use NSF channels controlled under the LUA program. In addition, 49 of the 220 fresh GNF2 reload bundles use NSF channels controlled separately under the LUC program, and 171 GNF2 reload bundles use Zircaloy-4 channels. The current COLR containing the operating limits and stability region boundaries and the reference core loading pattern (RLP) is revised to reflect operation of the Cycle 19 core.

During RFO-18 refueling, a failed (cladding failure) fuel bundle (RGE896) was removed from the core. The existence of the fuel failure had previously been identified in CR-RBS-2015-00009. This bundle was originally planned for use in the C19 RLP. Therefore, an Updated Loading Pattern (ULP) was prepared to complete the C19 core. Since the RLP configuration was used as the basis for the fuel licensing calculations, an evaluation performed on the ULP concluded that all licensing analyses remain applicable to the updated Cycle 19 core loading. This segment of the RBS design change was performed under the existing approved methods and required no further licensing evaluation; it was included here for completeness of the RBS C19 loading pattern description.

FO-16-01 was implemented to remove multiple fuel leakers from the RBS C19 core. The fuel leakers were replaced with fuel previously discharged to the Spent Fuel Pool (SFP) in RBS RF18, resulting in another Updated Loading Pattern (ULP). To distinguish between the two ULP designs, the first design from beginning of cycle (BOC) conditions evaluated in EC55796 is referred to as C19A ULP, while the second design resulting from FO-16-01 and evaluated in EC62442 is referred to as C19B ULP. This evaluation addresses the C19B ULP changes for operation of the core. Note that the ULP revision to the BOC19 core was performed using approved methods referenced in the RBS TS 5.6.5. EC62442 documents that the RBS C19B ULP resulting from FO-16-01 continues to meet all required core design and licensing criteria. As described in the preceding PAD, this 50.59 evaluation was completed because the C19B core has been re-configured and to document the impact of proposed changes to the RBS Core Operating Limits Report (COLR) which reflect the ULP design.

During FO-16-01, three failed (cladding failure) fuel bundles (RGE729, SGE003, and SGE039) were removed from the core. The existence of the fuel failures had previously been identified in CR-RBS-2015-3252. Through a combination of in-core shuffles and insertion of three GNF2 bundles previously discharged to the SFP in RF18 (QGE530, QGE605, QGE607), a ULP was prepared to complete the C19B core. Each of the re-insert bundles taken from the SFP was reviewed to reload criteria from EN-NF-105, "Reload Process", including visual inspection of each to confirm no debris had been captured in the upper or lower tie plates. Note that each individual bundle replacement in the ULP configuration (shuffle and SFP re-insert) was designed to ensure that

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

location in the C19B core would operate with a lower energy bundle than in the previous design. Since the C19A configuration was used as the basis for the fuel licensing calculations supporting the current RBS C19A COLR, an evaluation was performed on the C19B ULP; which concluded that all licensing analyses remain applicable to operation of the RBS C19B core. This segment of the RBS C19B design change was performed under the existing approved methods and demonstrated that the previous licensing evaluation remained applicable for operation of the C19B ULP.

Summary of Evaluation: The RBS C19 core design has been developed using NRC approved methods, General Electric Standard Application for Reactor Fuel (GESTAR-II), which are referenced in TS 5.6.5. The C19 core will be operated in conformance with cycle specific, generic, and fuel design specific analyses provided by Global Nuclear Fuel (GNF) using the GESTAR method. The C19 core operating limits have been incorporated into the core monitoring system prior to startup of C19. LAR-2015-01 was initiated to implement the required changes to the C19 COLR.

The ULP to reconfigure the RBS *BOC19* core without the failed fuel bundle was prepared using Entergy reload procedure EN-NF-105 *(EC55796)* and by NRC approved vendor methods (GESTAR-II). Through these methods, it was demonstrated that the revised core loading continued to meet the acceptance criteria of the reload licensing analyses without the need for any changes to the established C19 core operating limits. As such, no further licensing review was required.

An Enhanced Lead Use Channel program (per GNF Document MFN 12-074, Enhanced LUC Program for NSF Channels) has been implemented in RBS C19 to introduce the use of GNF's NSF alloy for fuel channels. RBG-47552 documents the notification to NRC of the LUC program implementation at RBS. A total of 8% (49 channels) of the C19 core fuel inventory uses the NSF LUCs, all with a pre-oxidized surface finish. This is part of a continuing effort to improve the resistance to distortion of the fuel channels used at RBS. Monitoring and inspection, per the NRC SER on MFN 12-074 (see Attachments 7.012 – 7.014 of EC55158), will be implemented in RF19 and later cycles to confirm the performance of the NSF LUCs. The mechanical characteristics of the NSF LUCs (including yield and tensile strength, ductility, corrosion resistance, etc.) are very similar to those of the Zircaloy-4 channels currently in use at RBS. The NSF channel material is being introduced as a GESTAR-II Lead Use application. The NSF LUC has extensive industry experience with exposures of channels up to 3 & 4 two-year cycles at several BWR units (including BWR/6). Monitoring and inspection results for these applications have demonstrated favorable performance.

A LUA program has been implemented in RBS C19 to introduce use of the GNF3 fuel assembly. RBG-47551 documents the notification to NRC of the LUA program implementation at RBS. A total of 4 GNF3 LUA were included in the RBS C19 core. The LUA is a part of a continuing industry effort to expand the operating experience with new fuel designs (improved performance in fuel efficiency). Monitoring and inspection of the GNF3 performance will be implemented in RF19 and later cycles. Under its LUA classification in the GESTAR-II method, the GNF3 fuel has been designed such that its mechanical and nuclear characteristics were confirmed to meet the required design acceptance criteria. GNF3 design characteristics are very similar to those of the GNF2 design already in operation at RBS. The GNF3 fuel will be operated in conformance with cycle specific, generic, and fuel design specific analyses provided by Global Nuclear Fuel (GNF) using the GESTAR method. The RBS C19 core was designed such that the GNF3 will operate with margin to the specified operating limits, and such that the GNF3 LUA will not be limiting at any time in C19. The GNF3 specific C19 core operating limits are incorporated into the core monitoring system prior to startup of C19. These fuel specific operating limits are addressed in LAR-2015-01 as a part of the changes to the C19 COLR.

The RBS C19B ULP core design (FO-16-01) has been developed using NRC approved methods, General Electric Standard Application for Reactor Fuel (GESTAR-II), which is referenced in TS 5.6.5. The C19B ULP core will be operated in conformance with cycle specific, generic, and fuel design specific analyses provided by Global Nuclear Fuel (GNF) using the GESTAR method. EC62442 documents that the C19A core operating limits continue to meet all core design and licensing criteria, and thus no changes are required for the current COLR operating limits. The only COLR change required is an update to one of its references; which cites the vendor evaluation of the ULP. The change in core inventory will be incorporated into the core monitoring system prior to startup of the C19 ULP (C19B). LAR-2016-01 was initiated to implement the required changes to the C19B COLR.

The C19A ULP reload core design included 220 fresh GNF2 bundles and 4 fresh GNF3 LUA, fabricated by Global Nuclear Fuel (GNF), were loaded during RFO18. SE #2015-001, Rev.0, OSRC Meeting 15-004 documented the introduction of the GNF3 Lead Use Assemblies (LUA) and of the GNF NSF Enhanced Lead Use Channels (LUC). The RBS C19B ULP does not impact the LUA or LUC components or programs as described in SE #2015-001, Rev.0, OSRC Meeting 15-004; thus the C19A ULP SE continues to apply to operation of the RBS C19B ULP. The C19B ULP to reconfigure the core without the failed fuel bundles was prepared using Entergy reload procedure EN-NF-105 and by NRC approved vendor methods (GESTAR-II). Through these methods, it was demonstrated that the revised core loading continued to meet the acceptance criteria of the reload licensing analyses without the need for any changes to the established C19 core operating limits. The current COLR was revised to reflect operation of the C19B ULP core. Note that the only change concerns an update to COLR Reference 3.1.5.

The reload licensing basis for the RBS C19 core is described in Ref.3.1.1 of the RBS C19 COLR, with evaluations of deviations in core design provided in COLR Ref.3.1.5. The latest update to COLR Ref.3.1.5 documents the fuel vendor reload licensing evaluation of the changes made with the RBS C19B ULP. The procedure for evaluating any core design deviations is part of the NRC approved reload licensing methodology, GESTAR-II, Section 3.4.2, "Acceptable Deviation from Reference Core Design". If any of the acceptance criteria in Section 3.4.2 are exceeded, the vendor must re-examine all the affected licensing parameters to ensure there is no adverse impact per GESTAR-II Section 3.4.3, "Re-examination Bases". The C19B core design met all of the Section 3.4.2 criteria with the exception of 'Locations of Fresh Reload Bundles'. Because two of the failed bundles were fresh reload assemblies from interior core locations and were replaced with other than fresh bundles, C19B required a full re-examination of the licensing parameters supporting the operating limits reported in the COLR Ref.3.1.1. The vendor evaluation in COLR Ref.3.1.5 includes re-examination of the following 8 parameters.

1,2) Scram Reactivity Insertion and Dynamic Void Coefficient

For Pressurization transients, the scram reactivity insertion was not degraded relative to the licensing basis for EOC & MOC conditions in the C19B ULP core configuration. The dynamic void coefficient was confirmed to be within the range analyzed for the licensing basis of EOC & MOC conditions. Therefore, the pressurization event Operating Limit Minimum Critical Power Ratio (OLMCPR) results for the core-wide Anticipated Operational Occurrence (AOO) reported in the COLR remain valid for the C19B ULP core.

In the RBS C19A design, all Thermal Over-Power (TOP) and Mechanical Over-Power (MOP) results for the Loss of Feedwater Heating (LFWH) event were within design limits and the LFWH event did not set the required OLMCPR. Due to small margin in the LFWH TOP results (with respect to setting the OLMCPR), the vendor ran confirmatory evaluations for the C19B ULP core configuration. The results indicated that the previous TOP/MOP results for the LFWH event remain valid for the C19B ULP.

3) Peak Fuel Enthalpy during Rod Drop Accident

Per the GESTAR-II methods, since RBS is a 'banked position withdrawal sequence plant', the control rod drop accident analysis is not required.

4) Cold Shutdown Margin

The predicted shutdown margin for the C19B ULP at FO-16-01 exposure (6279 MWd/ST) is greater than BOC19 (2.8%), which meets the design limit minimum value of 1%. The minimum shutdown margin for the cycle occurred at BOC19. A shutdown margin surveillance test (STP-050-3601, "Shutdown Margin Demonstration") is conducted at startup (at BOC19A and startup of C19B) to demonstrate compliance with technical specifications.

5) Standby Liquid Control System (SLCS) Shutdown Margin

The minimum SLCS shutdown margin was calculated at 3% in the C19A design; which meets the design limit minimum value of 1%. This is a beginning of cycle value that increases throughout the cycle. Since the overall reactivity of the C19B ULP core is reduced from that of the C19A design, the previously determined minimum shutdown margin remains appropriate for the C19B ULP design.

6) Change in Critical Power Ratio due to a Mislocated Fuel Assembly

RBS meets the requirements (described in GESTAR-II) for classifying the fuel loading events (mislocated and misoriented) as an Infrequent Incident; therefore cycle specific loading event analyses are not required.

7) Rod Block Monitor Response to a Rod Withdrawal Error (RWE)

The C19A design demonstrated that all TOP/MOP results for the RWE event were within design limits and did not set the required OLMCPR. Due to small margin in the MOP results for the RWE (with respect to setting the OLMCPR), the vendor ran confirmatory evaluations for the C19B ULP core configuration. The results indicated that the previous TOP/MOP results for the RWE event remain valid for the C19B ULP.

8) Safety Limit MCPR (SLMCPR)

The Safety Limit analysis assumes core quadrant fuel bundle symmetry. The introduction of the two second burn bundles into the interior locations designed for fresh reload fuel in the C19A configuration created asymmetries. Exposure and reactivity asymmetries tend to improve the SLMCPR since they increase radial peaking. Increased radial peaking reduces the total number of rods susceptible to boiling transition, thus reducing the SLMCPR value. Therefore, the SLMCPR confirmation performed for the C19A design remains applicable for the C19B ULP.

In addition to these evaluations, the vendor also performed analysis of the limiting reload validation matrix (RVM) cases that are used to confirm the Enhanced 1A (E1A) stability solution installed at RBS. The limiting RVM case 6 was evaluated and found to meet the required acceptance criteria using the C19B ULP core configuration.

Based on these evaluations, using NRC approved methods, the vendor demonstrated that the results previously reported in COLR Ref.3.1.1 remain valid for operation of the C19B ULP core.

Background Information

Mechanical

The GNF2 fuel mechanical design has been reviewed for use at River Bend (50.59 Evaluation 2011-004, OSRC Mtg# 11-006). No unusual failure modes or increased failure frequency have been identified for this fuel design. This is the 3rd cycle of operation for GNF2 fuel and this fuel design has accumulated operational experience at RBS and other plants (including Entergy BWR units) with no significant problems. The bundles will operate within the power history assumptions in the fuel mechanical analyses and will experience exposures within the analyzed burnup limits of the mechanical designs, including those bundles that will be irradiated for a third cycle. Mechanical design analyses [USAR 4.2.1] have been performed with NRC-approved methodology to evaluate mechanical criteria including cladding steady-state strain and stresses, transient strain and stresses, fatigue damage, creep collapse, corrosion, hydrogen absorption, fuel rod internal pressure, etc. All parameters were found to meet their respective design limits for the Cycle 19 core. Although an increased channel bow condition can result in increased friction between the control blade and its corresponding fuel assemblies, control rod settle and insertion testing (OSP-0061) will continue to be performed during Cycle 19 per the guidance provided in fleet procedure EN-RE-216 to ensure that any increased axial friction loads on the channel and fuel assembly load chain remain below acceptable limits. Numerous previously irradiated C19 fuel assemblies (total of 31) were re-channeled with new Zircaloy-4 channels (identical replacements) to enhance the C19 core in terms of its resistance to channel distortion (i.e., reduce potential for channel bow).

The ULP to reconfigure the RBS C19 core without the failed fuel bundle was prepared using Entergy reload procedure EN-NF-105 and by NRC approved vendor methods (GESTAR-II). Through these methods, it was

demonstrated that the revised core loading continued to meet the fuel mechanical acceptance criteria of the reload licensing analyses without the need for any changes to the established C19 core operating limits.

The mechanical characteristics of the NSF LUCs (including yield and tensile strength, ductility, corrosion resistance, etc.) are very similar to those of the Zircaloy-4 channels currently in use at RBS. Use of NSF LUC has been previously addressed for operation in the GGNS C20 core and found acceptable (SE-2014-001). The mechanical performance of NSF channels for GNF2 fuel is bound by that of channel designs previously used at RBS, such as Zircaloy-2 and Zircaloy-4.

The GNF3 mechanical design has been reviewed for use at RBS. No unusual failure modes or increased failure frequency have been identified for this fuel design. The bundles will operate within the power history assumptions in the fuel mechanical analyses and will result in exposures within the analyzed burnup limits of current fuel mechanical designs. The mechanical compatibility of the GNF3 LUA with co-resident GNF2 fuel in the RBS core has been reviewed and found acceptable in EC53147 & EC54888 (fuel receipt), in EC55250 (fuel shuffle), and EC55796 (C19 Reload). The GNF3 channels are fabricated using the NSF alloy, but are controlled under the LUA program and exclusive from the NSF LUC program. The GNF3 bundle and channel design was reviewed in EC55796 and found acceptable for operation.

The GNF3 bundles were designed using approved vendor methods for LUA design in GESTAR-II (referenced in RBS TS 5.6.5), and the methods are capable of analyzing all aspects of the new fuel design. The GNF3 LUA have been analyzed for RBS C19 specific operations such that they will not be the most limiting fuel assemblies at any time during C19 operation with respect to compliance with fuel mechanical acceptance criteria. Such analyses shall be performed for each subsequent cycle of GNF LUA operation in the RBS core to confirm continued operations in this manner.

Nuclear

The neutronic characteristics of the Cycle 19 core design have been considered in the safety analysis. The nuclear characteristics of the reload GNF2 fuel remain unchanged from previous RBS reloads. Since the ULP design changes were performed in accordance with approved methods and only involve previously irradiated GNF2 fuel, there will be no impacts on the nuclear response of the core. The nuclear characteristics of the NSF LUCs are very similar to that of currently used Zircaloy-4 fuel channels, no nuclear related impacts are expected from introduction of the NSF LUCs. The nuclear characteristics of the GNF3 fuel are similar to those of the GNF2 bundles they displaced in the RBS C19 reload fuel. Their placement in the core was analyzed to ensure they will not be the most limiting fuel assemblies in the core at any time during operation of C19. Such analyses shall be performed for each subsequent cycle of GNF LUA operation in the RBS core to confirm continued operations in this manner. Adequate shutdown margin has been predicted by analysis and will be confirmed during startup tests per Technical Specification 3.1.1. In addition, the hold-down capability of the standby liquid control system has been confirmed. Therefore, the probability of inadvertent criticality has not been increased by the introduction of GNF3 LUA and NSF LUC in the Cycle 19 reload fuel.

Channel General and Safety Related Design Functions

The fuel channel component includes general design functions and safety related design functions.

The general design functions include:

1. The channel forms the flow path shell for fuel bundle coolant flow.
2. The channel provides surfaces for control rod guidance in the reactor core.
3. The channel provides structural stiffness to the fuel bundle during lateral loadings applied from fuel rods through the fuel spacers
4. The channel forms the coolant flow leakage path at the channel/lower tie plate interface.

5. The channel transmits fuel assembly seismic loadings to the top guide and fuel support of the core internal structure.

The safety functions of the fuel channels include:

1. The four channels in a cell establish the pathway through which the control blade moves. The lateral stiffness of the channel prevents channel buckling and ensures that the pathway remains available during design basis events, such as an earthquake. Excessive bulge and bow of channels may affect the movement of rods and the scram time. Therefore, the fuel channels impact the capability to shut down the reactor and maintain it in a safe shutdown condition.

2. The channel provides a barrier to allow parallel coolant flow paths and provides a heat sink which cools the outer row of fuel rods during a LOCA event.

3. The channel provides a barrier to fuel rod failure propagation from one fuel assembly to others by maintaining separation of fuel rods and by restricting the movement of debris that may be associated with the initial failure. Therefore, the fuel channel may help to mitigate the consequences of certain severe accidents.

The NSF alloy has demonstrated improved bow characteristics in prior lead use programs. Hence, the operational experience in these lead use programs suggests that NSF will perform with greater resistance to the various forms of channel distortion (i.e., fluence gradient-induced bow, shadow corrosion-induced bow) and have corrosion performance similar to the current Zircaloy-4 channels in the RBS core. The previous LUC programs provide a degree of confidence that there will be no unanticipated issues in the behavior and performance as compared to Zircaloy. The GNF3 NSF channel design was reviewed and found acceptable for operation in the RBS core in EC55796.

The material properties of the NSF alloy are sufficiently similar to current channel materials such that NSF channels are expected to be dimensionally, structurally and mechanically compatible with Zircaloy-4 channels within a four bundle cell. Differences in the mechanical, thermal, and nuclear properties of NSF, as compared to Zircaloy-4, are minor and do not affect the methodology used to design and analyze the channels. Similarly, the minor alloy differences between NSF and Zircaloy have little to no significance to core design, transient, stability or ECCS-LOCA analyses.

Is the validity of this Evaluation dependent on any other change? ☐ Yes ☒ No

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ Yes ☒ No

Preparer: Jim Head / See AS* / ESI / Fuels Analysis / 1/25/2016
Name (print) / Signature / Company / Department / Date

Reviewer: W. Steelman / See AS* / EOI / Licensing / 1/25/2016
Name (print) / Signature / Company / Department / Date

OSRC: Dean Burnett / Dean B / 1/25/2016
Chairman's Name (print) / Signature / Date

16-00X

OSRC Meeting # 2016-001

* EC 0000062442

II. 50.59 EVALUATION

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. **Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR?** ☐ **Yes** ☒ **No**

BASIS:

RBS Cycle 19 operation will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR. The precursors to these events are independent of the core design and the frequency classifications reported in USAR Chapter 15 are unaffected by the core parameters. The following considerations support this conclusion.

Core Operating Limits

Operating Thermal Limits

Having followed approved design methods for the *RBS C19B core design*, power operation within the core operating limits (Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR), Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)) will ensure that the appropriate safety criteria are met during normal operations, Anticipated Operational Occurrences (AOO), or accidents. The precursors to these events are independent of the core design and the frequency classifications reported in SAR Chapter 15 are unaffected by the core parameters. Therefore, the core parameters in the COLR have no effect on the probability of the occurrence of any accident described in the SAR.

APRM Limits and Stability Region Boundaries

As with the normal reload review process for any core design, the stability performance of the reload design must be evaluated against various acceptance criteria specified in the E1A stability solution. Analyses have been performed to demonstrate that the revised E1A region boundaries for Cycle 19 met the channel and core decay ratios acceptance criteria. The change in E1A stability region boundaries implemented in the RBS C17 reload design will continue to avoid a thermal-hydraulic instability during the Cycle 19 operation. The *RBS C19B ULP core design* with NSF LUC and GNF3 LUA has been analyzed and validated to meet all required stability performance criteria. Therefore, the probability of a thermal hydraulic instability has not increased.

Design Basis Accidents (Limiting Faults)

The design basis accidents analyzed in support of Cycle 19 are the control rod drop accident (CRDA), the fuel handling accident (FHA), and the loss of coolant accident (LOCA). The FHA is discussed separately.

The probability of the occurrence of design basis accidents is not dependent on the core configuration. No changes to the plant design are required for the *Cycle 19B core*. The *Cycle 19B core loading* will not affect the precursors to any of the Chapter 15 events, including LOCA analyses.

The Control Rod Drop Accident (CRDA) results from a failure of the control rod-to-drive mechanism coupling after the control rod becomes stuck in its fully inserted position. Although a channel bow condition can result in friction between the control blade and its corresponding fuel assemblies, analyses

have shown that there would not be sufficient friction to result in a mechanical failure of the coupling. Additionally, the control rod drive mechanism would not produce enough force to result in a mechanical failure of the coupling even if the channel bow was so severe that the assemblies would preclude blade movement. Channel bow associated with GNF2 reload fuel at high exposure is no more than the channel bow associated with previous RBS reload fuels (i.e., ATRIUM-10, GE14). The addition of NSF LUCs and the GNF3 LUA is not expected to have any negative impact on channel performance in the core; based on industry experience with other LUC programs with multiple cycles of exposure, the performance of the NSF LUCs has been similar to or improved over that of the Zircaloy-4 channels already present in the RBS core. As such, channel bow is not considered a precursor to the CRDA, and any increased bow associated with the high exposure fuel bundles would not increase the probability of this event.

Fuel Handling Accident and Fuel Loading Error

The FHA related impacts of GNF2 were previously evaluated for RBS application in RBS C17. *Since the RBS C19B ULP makes use of re-insert GNF2 fuel from the same RBS C17 reload batch fuel, the FHA related impacts from use of GNF2 continue to apply.* As described in RBS C19 Fuel Receipt EC54888, the GNF3 design is very similar in its overall physical characteristics to that of GNF2 and that its interface with handling equipment is likewise similar to GNF2, there is no expectation for an increase in the probability of occurrence of a FHA.

The fuel loading error (FLE) event involves improper loading of a fuel bundle into the core, related to human error associated with the ability to properly identify a bundle by its serial number identification and ability to determine the proper orientation of the bundle when loading into the core. These factors are driven by design features of the upper tie plate of the bundle, where the bundle ID is machined and orientation queues are built into the UTP design. The identification and orientation features of the GNF3 design are consistent with that of GNF2. Thus, there is no increase in the probability of occurrence of an FLE event.

On these bases, the probability of occurrence of accidents previously identified in the SAR is not increased for the Cycle 19B core.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR? ☐ Yes ☒ No

BASIS:

No plant modifications are required to accommodate the Cycle 19 core design. Related specifically to the fuel in the reloaded core, the only additional loads placed on plant equipment would be due to changes in the characteristics of the fuel as it interfaces with other core components and functions; such as stability, decay heat, reactor internal pressure differences, reactor internal structural impacts, transient responses, neutron fluence, radiation source term, recirculation system impacts, changes to EOPs, fuel storage criticality, or other severe accident related affects. All of these topics were specifically addressed by the fuel vendor's assessment for introduction of the GNF3 LUA design into the RBS core. All areas of review found acceptable results; with either no impact identified or minor changes that are within the current RBS design and licensing basis limits for fuel and core design (i.e., GNF3 accounted for in Thermal Limits results applied in the C19 COLR, GNF3 design part of the bases for revised RBS SFP Criticality analysis, etc.). The C19 reload evaluation has shown that the GNF3 LUA are compatible with co-resident fuel and core components in terms of its form, fit, and function. GNF3 does not introduce an increased potential for interference with core components (such as control rod blades, instrument tubes, etc.). Therefore, GNF3 introduction into the RBS C19 core does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of core components.

The only additional loads placed on plant equipment associated with the reload GNF2 fuel would be due to increased friction between the control blades and excessively bowed channels. As described in EC55796, the ULP design includes fuel channel designs previously used at RBS, and the re-channel of numerous

previously irradiated fuel bundles in the RBS C19 core will reduce the potential for occurrence of excessive channel bow. *As described in EC62442, the C19B ULP design likewise continues to use previously proven channel designs. The RBS C19B ULP design has been analyzed for the potential to develop excessive channel bow and found acceptable.* Introduction of NSF LUCs does not impact previously evaluated SSC's or their malfunction as described in the USAR since its properties are very similar to those of the co-resident Zircaloy-4 fuel channels in the RBS core. Based on previous experience with bowed fuel at RBS and other BWR-6 units, increased control blade friction can result in increased control rod settle times but is not expected to significantly impact scram times. Technical Specification scram time testing and control rod settle and insertion testing will continue to be performed during Cycle 19, in accordance with guidance provided in fleet procedure EN-RE-216. These actions would identify any potential scram time or other impacts such that appropriate corrective actions are taken. These actions will ensure that any increased control blade friction loads are not sufficient to cause any failures associated with the control blades or the control blade drive system, the fuel assembly load chain, or the vessel internals. Regarding NSF LUC conditions and requirements specified by NRC in their SER of the Enhanced LUC program, additional monitoring (settle testing and scram time testing) and inspections must be performed to confirm the performance of the LUCs. This additional monitoring and inspection will begin in RFO19 and for operation of the NSF LUC in RBS Cycle 20 and beyond.

Analysis of Reactor Internal Pressure Differences (RIPDs) resulting from the introduction of a new fuel type with different hydraulic characteristics is done to ensure design basis criteria are met. The pressure across the RBS C19 core is the result primarily of the dominant fuel in the core, GNF2.The introduction of four GNF3 LUA in the 624 bundle RBS core has very little impact on the overall core hydraulic characteristics. In addition, the GNF3 hydraulic performance is bounded by past fuel types used at RBS. *The changes introduces with the RBS C19B ULP design continue to have the same hydraulic performance characteristics as found in the C19A ULP design (i.e., each core location has the same fuel design type in both the C19A and C19B ULP cores).*

A reactor internal structural assessment has been performed for the reload fuel. This assessment included normal, upset, emergency and faulted condition loads on the reactor internals and concluded that the reactor internals remain qualified for the reload fuel. The evaluation of GNF3 LUA found no adverse impacts to the structural integrity of the reactor internals with regard to RIPD or seismic loading. The GNF3 assemblies have an insignificantly greater weight than GNF2. The loading plan for the GNF3 LUA places one bundle in each of the four quadrants of the RBS core, thus avoiding any local concentration of the added weight. When analyzed on a cell basis, the additional weight averaged over a 4-bundle cell with 3 co-resident GNF2 assemblies is approximately 0.5%, which is considered negligible relative to structural integrity. The magnitude of the increase in bundle weight of GNF3 versus GNF2 is comparable to past RBS reloads of mixed fuel types such as GE14 versus ATRIUM-10. Based on evaluating the combined effects of RIPD, seismic, and fuel assembly weight, the introduction of four GNF3 LUA will have an insignificant effect on the structural integrity of the reactor internal components.

The dominant fuel type, GNF2, dictates the seismic behavior if the RBS fuel core response. Introduction of the four GNF3 LUA does not have any significant impact on the mass and stiffness of the RBS C19 fuel core, and does not change the fuel fundamental frequency. The GNF3 fuel lift margin and control rod guide tube lift forces are bounded by previous fuel types used at RBS. Thus the design basis fuel lift loads remain valid for the four GNF3 LUA. The seismic capability of the GNF3 spacers was confirmed through seismic load testing. *The changes introduced with the RBS C19B ULP maintain the same ratio of GNF2 to GNF3 inventory as the C19A design.* The current design seismic qualification of the RBS C19 core in supporting operation with four GNF3 LUA remains unchanged

A conservative vessel over-pressurization analysis has been performed, which shows that the vessel pressure limit is not exceeded. The results of the analysis were reported in the Supplemental Reload Licensing Report (SRLR) for the RBS C19 core, in which the GNF3 LUA was explicitly modeled. The RBS C19 core with GNF3 LUA was shown to continue to meet the required ASME overpressure protection criteria. *The evaluation performed in support of EC62442 found that the RBS C19 SRLR results remain unchanged by the RBS C19B ULP design.*

The impact on the vessel fluence associated with the RBS C19 reload was evaluated in EC55796. It was found that the current RBS vessel fluence analysis remains valid with the C19 reload core design. Given the interior location of the GNF3 LUA in the RBS C19 core with respect to the vessel wall, GNF3 does not impact vessel fluence in operation of the C19 core and the current RBS P/T curve remains applicable. Thus GNF3 does not result in an increase in the likelihood of occurrence of fluence induced damage to the vessel. *The evaluation performed in EC62442 found that the RBS C19B ULP design does not impact the vessel fluence; the current RBS vessel fluence analysis remains valid with the C19B reload core design.*

The containment response is determined primarily by fuel-independent parameters such as RHR heat exchanger performance, ECCS coolant injection flow, service water temperature, and initial suppression pool temperature. As previously described, introduction of the GNF3 LUA does not result in more than a minimal impact on the containment response.

With regards to post-LOCA hydrogen control, per 10CFR50.44 RBS must have a system that will mitigate hydrogen from a metal-water reaction involving 75% of the core cladding surrounding the active fuel region. The containment igniter system meets this requirement for the current GNF2 reload core. The addition of GNF3 LUA results in a bundle specific comparison of approximately 3% increased cladding surface in the active fuel region (i.e., GNF3 has approximately 3% more cladding surface in the active fuel region than GNF2). However, comparison of the total active surface area in the C19 core configuration shows that it is bounded by that of the C18 configuration. *The ratio of GNF2 to GNF3 assemblies in the RBS C19B core is identical to that of the RBS C19A design.* Therefore, introduction of the GNF3 LUA does not have more than a minimal impact post-LOCA hydrogen control.

The precursors to any malfunction of equipment important to safety are not affected by the Cycle 19B reload core. Therefore, there is not more than a minimal increase in the likelihood of an occurrence of a malfunction of a SSC important to safety previously evaluated in the USAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR? ☐ **Yes** ☒ **No**

BASIS:

The acceptance criteria reported in USAR Section 15.0.3.1 and the Technical Specifications are satisfied for each event classification. Core operating limits have been developed to ensure that moderate frequency events do not violate the MCPR safety limit or fuel cladding strain limits. The consequences of infrequent events have been shown to meet the appropriate acceptance criteria while the individual acceptance criteria for the limiting faults have been demonstrated to be satisfied. Satisfying these acceptance criteria ensures an upper bound on the potential release of radiological source term, thus limiting the resulting consequences of the event. As such, the consequences of infrequent events and limiting faults described in the USAR are unchanged for the Cycle 19 reload core. The following considerations support these conclusions.

Anticipated Operational Occurrences (AOO) Events

The Cycle 19 core operating limits have been developed with NRC-approved methodologies such that the MCPR safety limit and the fuel cladding strain limit will not be violated by any analyzed moderate frequency transient initiated from any statepoint available to RBS. *The evaluation performed in support of EC62442 found that the RBS C19 core operating limits (including stability region boundaries) remain unchanged by the RBS C19B ULP design. Fuel vendor analysis of the C19B core design included confirmatory evaluations of the Loss of Feedwater Heating (LFWH) and Rod Withdrawal Error (RWE) events. In addition, the limiting reload validation cases were evaluated for the RBS E1A stability regions. In every evaluation, the C19B results remained within design limits,*

confirming that the existing licensing conclusions presented in RBS COLR Ref.3.1.1 remain valid for operation of the C19B core. As such, no fuel failures are expected to result from any moderate frequency event. Thus the radiological source term and the potential extent of its release from the fuel have not been increased with the introduction of GNF3 LUA in the RBS C19 core. These analyses considered RBS-specific operational modes such as Maximum Extended Operating Domain (MEOD), Single Loop Operation (SLO), Feedwater Heater(s) out of Service (FHOOS), Final Feedwater Temperature Reduction (FFTR), Pressure Regulator Out Of Service (PROOS), Main Turbine Bypass Out Of Service (MTBOOS) and End Of Cycle –Recirculation Pump Trip (EOC-RPT) inoperable. These core operating limits consist of MCPR and LHGR curves that are functions of flow, power, and exposure for both GNF2 and GNF3 fuel. These core operating limits have been incorporated into the core monitoring system. These limits consider conservative channel bow assumptions that bound the current measured bow data and the expected increased bow associated with the highly exposed fuel. Introduction of the NSF LUC is not expected to impact the core with respect to performance of the fuel channels. Industry experience with other applications of NSF LUC has yielded favorable results. GNF3 LUA have been loaded into the RBS C19 core such that they will operate with margin to the thermal limit requirements. Approved methods were used to model the GNF3 LUA and confirm they respond in a similar manner to the GNF2 assemblies they replace in the RBS C19 core. Thus introduction of the GN3 LUA and NSF LUC to the RBS C19 core design does not result in an increase in the consequences any AOO event.

Infrequent Events

The fuel loading error (*i.e.,* misoriented and mislocated) is considered an infrequent event. The consequence of this event has been evaluated in accordance with the GESTAR requirements and shown to meet the respective acceptance criteria. Radiological analyses using the alternative source term (AST) have been performed to ensure that these events will not result in an increase in offsite or control room dose or doses greater than their respective acceptance criteria. Evaluation of the *RBS C19B core design in EC62442*, including GNF3 LUA and NSF LUC demonstrated that the current radiological source term for RBS remains bounding. The radiological analyses reflect operation as described in the USAR.

Limiting Faults

The limiting faults at RBS include the fuel handling accident, the control rod drop accident, and the design basis LOCA. The radiological analyses for these events have been developed as part of the AST effort assuming certain core parameter values such as radial peaking factor and fuel burnup. Core parameters assumed in the dose analysis were reviewed and found to be bounding for the Cycle 19 core. The number of rod failures in a CRDA or FHA using GNF2 fuel remains bounded by the assumption used in the dose analyses. The Cycle 19 core (with inclusion of the GNF3 LUA) parameters have been shown to be bounded by the RBS AST source term and radiological analyses. For the Loss Of Coolant Accident (LOCA), MAPLHGR operating limits and single-loop multipliers have been developed for the Cycle 19 core configuration, including GNF3 LUA, such that the requirements of 10CFR50.46 are satisfied. The containment response for the Cycle 19 core, including GNF3 LUA, was found to not result in more than a minimal impact on previous GNF2 design based containment system response analysis. Review of the potential degraded core hydrogen generation has shown that the RBS C18 core configuration remains bounding with respect to the Cycle 19 core configuration, including GNF3 LUA. *The evaluation performed in EC62442 found that the RBS C19B ULP design does not impact the current AST or radiological analyses; the current RBS source term and dose analyses remain valid with the C19B reload core design. The ratio of GNF2 to GNF3 assemblies in the RBS C19B core is identical to that of the RBS C19A design.* Review of the Cycle 19 core, including GNF3 LUA, has shown that there will not be more than a minimal impact on the seismic/LOCA response. The GNF3 LUA design has no effect on the LOCA and Transient Analyses and the consequence from these events is not impacted.

Compliance with licensing acceptance criteria for the CRDA is assured through adherence to the Banked Position Withdrawal Sequence (BPWS) where analyses have generically demonstrated large margin to licensing limits on enthalpy insertions. Given the similarities of the nuclear characteristics of the GNF3 and GNF2, the generic analysis remains applicable to RBS operation with GNF3 LUA.

With respect to LOCA performance of the RBS C19 core, the GNF3 bundles will be restricted to a peak linear heat generation rate (PLHGR) equivalent to that of GNF2. The GNF3 licensing basis peak clad temperature (LBPCT) was determined using approved methods and demonstrated conservative margin to the 10CFR50.46 PCT limit. As previously discussed, the change in post-LOCA hydrogen produced from a metal-water reaction with the GNF3 fuel in the C19 core configuration remains bounded by that from the C18 configuration. Since the LBPCT and maximum local oxidation remain within licensing limits, a coolable geometry is assured. Likewise, introduction of GNF3 does not impact the reflooding capability of the ECCS or operation of the core spray systems, thus also does not impact long term cooling. Therefore, the acceptance criteria of 10CFR50.46 remain satisfied with introduction of the GNF3 LUA.

The FHA was analyzed previously for introduction of GNF2 in RBS C17. *Since the RBS C19B ULP makes use of re-insert GNF2 fuel from the same RBS C17 reload batch fuel, the FHA related impacts from use of GNF2 continue to apply.* The analysis was performed for GNF3 bundles in the RBS core design. Given that the GNF3 LUA are loaded in non-limiting locations of the C19 core, they will accumulate less bundle exposure and less bundle source activity. Despite this loading restriction, the analysis applied the same conservative assumptions regarding exposure and radial peaking factor, which increased the assumed bundle source activity more than that associated with the slight increase in fuel mass of GNF3 versus GNF2 bundles. The results of the FHA analysis demonstrated that even given these conservative assumptions, the consequence multiplier of the GNF3 fuel remains less than unity and is therefore bounded by the consequence of a FHA involving GNF2 fuel, confirming that the GNF3 LUA have no impact on the dose consequences of the FHA.

The core radiation source term impact from the introduction of GNF3 is inconsequential since its design is similar to that of GNF2. As previously described, the C19 reload parameters affecting the radiological source term were compared against the acceptance criteria and found to remain bounded. Therefore, GNF3 LUA in the C19 core are expected to have no significant impact on the RBS core radiological source term. Radionuclide concentrations in reactor water and steam are not dependent on fuel type but on operating conditions of the core, and since GNF3 will not affect core performance; it will not affect coolant radiation sources. Therefore, GNF3 will not affect the dose consequences of postulated reactor coolant system accidents such as Main Steam Line Break.

A recirculation pump seizure event while in single loop operation is also a limiting fault, but is conservatively evaluated against the AOO acceptance criteria. As the AOO acceptance criteria (MCPR) are satisfied, the limiting fault acceptance criteria are satisfied. This was demonstrated for the introduction of GNF2 in RBS C17. *Since the RBS C19B ULP makes use of re-insert GNF2 fuel from the same RBS C17 reload batch fuel and continues to meet the AOO acceptance criteria (using the same MCPR limits as the C19A design), the limiting fault acceptance criteria are satisfied.* The introduction of four GNF3 LUA into RBS C19 will not affect the results of the analysis due to the large conservatisms assumed in the calculation of the void coefficient. For the GNF3 LUA, the same critical power correlation as used for GNF2 (GEXL17) has been conservatively adjusted for application to GNF3. Therefore, the previously determined GNF2 SLO pump seizure limits apply to GNF3.

Based on the review above, it is concluded that the Cycle 19 reload design will not increase the consequences of an accident previously evaluated in the USAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR? ☐ **Yes** ☒ **No**

BASIS:

All equipment important to safety will function in the same manner with the *Cycle 19B ULP core* as with the previous core design. The proposed reload does not require new hardware or modification of existing hardware. Any malfunction of key plant components as described in USAR which could impact fuel integrity has been factored into the Cycle 19 transient and accident analyses. The consequences of these malfunctions have been shown to meet their respective acceptance criteria and to remain unchanged and

bounded for Cycle 19 operation. Specific to the Cycle 19 reload design with the introduction of GNF3 LUA, it has been shown to be compatible with the other co-resident GNF2 fuel in the core. GNF2 fuel was introduced at RBS in C17, and it has demonstrated acceptable performance in reload evaluations and operations for C17 and C18. As previously described, the performance of GNF3 as determined using approved methods is predicted to be similar to that of GNF2. Introduction of the NSF LUC is compatible with the Zircaloy-4 channels currently used in the RBS core, and is expected to provide comparable or improved performance during operation. Specific topics are discussed below:

Core Operating Limits

The SLMCPR and AOO analyses are based upon Cycle 19 core design to determine operating thermal limits, using the NRC approved methodology. Operation within these limits during normal steady state operation provides assurance that MCPR safety limits and fuel thermal/mechanical design limit are not exceeded due to the most severe AOO event. An AOO event assumes a single equipment failure such as load reject or turbine trip which causes a spike in reactor power and heat flux. The transient analyses have included such severe events so that safety limits are always maintained. The applicability of the current stability region boundaries depicted in the COLR has been confirmed for Cycle 19. *The evaluation performed in support of EC62442 found that the RBS C19A core operating limits (including stability region boundaries) remain unchanged by the RBS C19B ULP design. Fuel vendor analysis of the C19B core design included confirmatory evaluations of the Loss of Feedwater Heating (LFWH) and Rod Withdrawal Error (RWE) events. In addition, the limiting reload validation cases were evaluated for the RBS E1A stability regions. In every evaluation, the C19B results remained within design limits, confirming that the existing licensing conclusions presented in RBS COLR Ref.3.1.1 remain valid for operation of the C19B core.* As such, no fuel failures are expected to result from any AOO event. The loading the GNF3 LUA has been designed such that they will not be the limiting fuel in the core at any time in C19 operations. The core loading for the GNF3 LUA has been designed such that the GNF3 will operate with margin to the operating limits. Thus the RBS C19 core is designed such that the potential release of bundle source radionuclides remains bounded by the limiting source term analysis. Therefore, the consequences of a malfunction of a safety-related structure, system or component have not increased.

Design Basis Accident

The design basis accidents reviewed in support of Cycle 19 are the control rod drop accident (CRDA), the fuel handling accident (FHA), and the loss of coolant accident (LOCA). The core parameters assumed in the RBS dose analyses remain bounding with respect to the Cycle 19 operating core parameters with introduction of GNF3 LUA. *The evaluation performed in EC62442 found that the RBS C19B ULP design does not impact the current AST or radiological analyses; the current RBS source term and dose analyses remain valid with the C19B reload core design.*

Therefore, Cycle 19 operation will not increase the consequence of a malfunction of equipment important to safety evaluated in the SAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the USAR? ☐ **Yes** ☒ **No**

BASIS:

The Cycle 19 GNF3 fuel has been shown compatible with GNF2 fuel that was inserted in previous cycles. There are no changes in the operation of any system; highly exposed fuel bundles are replaced with fresh bundles. The malfunction of key components as described in USAR are analyzed in the transient and accident analysis of the Cycle 19 core as part of the NRC approved reload analysis methods. Cycle 19 operation will be operated within the design limits to protect fuel integrity. No plant modifications are required to accommodate the new core design or to operate Cycle 19. *The evaluation performed in support of EC62442 found that the RBS C19 core operating limits (including stability region boundaries) remain unchanged by the RBS C19B ULP design.* As previously described, the design and operating features of the GNF3 LUA are very similar to that of GNF2, such that there are no new failure modes created with its introduction into the RBS C19 core. No additional operating modes are required for the

operation of Cycle 19. The loading of GNF3 LUA in the RBS C19 core design is such that they will operate with margin to thermal limits.

The introduction of NSF LUCs into the RBS core is not expected to result in any anomalous performance of the fuel channels; previous industry experience with NSF has shown corrosion resistance that is slightly improved over that of the Zircaloy-4 channels currently in the RBS core. The structural and physical properties of the NSF alloy is similar in terms of parameters such as stiffness and yield strength to that of the Zircaloy-4 alloy already in the RBS core. Thus its ability to perform the design function of a fuel channel (i.e., provides coolant flow path, guide for control rod travel, structural support for fuel in lateral loading, coolant flow leakage path at the core support plate which is important for in-core instrument function, transmits seismic loadings to the top guide and the fuel support/core internal structures, etc.) is similar to that of the fuel channels already present in the RBS core.

Therefore, the proposed activities will not create the possibility of an accident of a different type than any evaluated previously in the SAR.

6. **Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the USAR?** ☐ Yes ☒ No

BASIS:

The Cycle 19 reload core design cannot create the possibility of a malfunction of equipment important to safety with a different result than any previously evaluated. As previously described, the design and operating features of the GNF3 LUA are very similar to that of GNF2, such that there are no new failure modes created with its introduction into the RBS C19 core. Equipment important to safety will still be required to function in the same manner with the *Cycle 19B core* as with previous core designs. No plant modifications are required to accommodate the new fuel. The change in core characteristics does not change any parameter that would affect the function of equipment important to safety.

The *Cycle 19B ULP fuel* and GNF3 LUA are compatible with fuel that was inserted in previous cycles. The fuel will be operated within the design limits which will ensure all mechanical design criteria (e.g., centerline melt, clad strain) are met throughout the expected bundle lifetime. The introduction of the NSF LUCs is not expected to impact any of these design functions since the mechanical properties of the NSF alloy are very similar to those of Zircaloy-4. Previous experience at other BWRs (including BWR/6) of the NSF LUCs has demonstrated favorable performance over several cycles of continuous in-core exposure. Since the fuel designs meet all the fuel design acceptance criteria, the Cycle 19 core will not create a new mode of malfunction of equipment. Hence, introduction of the GNF3 LUA and NSF LUC into the Cycle 19 core will not increase the possibility of a malfunction of equipment important to safety with a different result than any evaluated in the SAR.

Therefore, Cycle 19 operation will not create the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the USAR.

7. **Result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered?** ☐ Yes ☒ No

BASIS:

Mechanical analyses have been performed to ensure that all fuel in the *Cycle 19B core* meets the mechanical design limits for steady-state operation as well as transient conditions including fatigue damage, creep collapse, corrosion, fuel rod internal pressure, rod bow, internal pressure, etc. This includes the GNF2 and GNF3 fuel. Additionally, no *Cycle 19B fuel* will exceed the applicable burn-up limits or thermal-mechanical integrity limits. The *C19B core design* ensures that the GNF3 LUA are operated with margin to the required operating limits.

Core operating limits have been developed using NRC approved methods to ensure that the *Cycle 19A fuel* will not exceed the MCPR safety limits for steady-state operation and Anticipated

Operational Occurrences (AOO). Similarly, operating limits have been developed to ensure that the *Cycle 19A fuel* will not exceed the 1% cladding strain limit or experience core-wide fuel melt during steady-state operation or AOO's. *As previously described, vendor evaluations documented in COLR Ref.3.1.5 demonstrate, using NRC approved methods, that the C19A operating limits remain valid for operation of the C19B ULP core.*

As described in EC55796, a bounding pressurization event with a failure of the direct scram has been analyzed for Cycle 19 to ensure compliance with American Society of Mechanical Engineers (ASME) code requirements. This analysis indicates that the vessel pressure safety limit is not exceeded for Cycle 19. *As described in EC62422, the RBS C19B ULP design continues to satisfy the ASME code requirements, indicating that the vessel pressure safety limit is not exceeded for Cycle 19B.*

A design basis limit for the peak fuel enthalpy of 280 cal/gm has been established for the control rod drop accident (CRDA) to preclude significant fuel cladding failure such that core geometry and cooling may be impacted. An evaluation has demonstrated that the generic GNF CRDA analysis is applicable to *RBS Cycle 19B core operation*, with GNF3 LUA. This generic analysis shows that a CRDA will not exceed the 280 cal/gm peak enthalpy limit. Since this accident is a localized event and the peak enthalpy does not exceed 280 cal/gm, there is no impact on the vessel or containment pressures. As such their respective limits are not exceeded.

10CFR50.46 provides limits associated with the ECCS performance analysis (LOCA analysis). Two such limits are Peak Clad Temperature (PCT) and local clad oxidation. Although these limits are not subject to 10CFR50.59, they are discussed in this evaluation for completeness. River Bend specific analyses have been performed for GNF2 and GNF3 fuel in accordance with 10CFR50.46. These analyses, which are applicable to Cycle 19, show that the PCT and local oxidation are well below the limits set forth in 10CFR50.46. The core-wide metal water reaction, which is used to evaluate compliance with the containment design limit, is bounded by the C18 configuration. The containment pressure design limit will not be exceeded in *Cycle 19B*. The GNF2 & GNF3 designs have no impact on LOCA performance of the fuel, as described by GNF in demonstrating compliance of the design with GESTAR-II licensing requirements.

An Anticipated Transient Without Scram (ATWS) evaluation was performed for GNF2 new fuel introduction at RBS in C17. As described in EC55796, this evaluation is applicable to Cycle 19 operation. *Since the RBS C19B ULP makes use of re-insert GNF2 fuel from the same RBS C17 reload batch fuel, the ATWS related impacts from use of GNF2 continue to apply to operation of Cycle 19B.* As previously described, the use of GNF3 fuel in the RBS C19 core was confirmed to not impact the existing ATWS core-wide analysis results. The fuel specific GNF3 ATWS results likewise remain well bounded by the ATWS event acceptance criteria. The evaluation demonstrates that the vessel pressure remains below the ASME emergency vessel pressure limit of 1500 psig and the containment parameters (suppression pool temperature, containment pressure) remain below applicable design limits for the ATWS event.

Additional cycle-independent evaluations have been reviewed or performed for Cycle 19 operation, including Appendix R (Fire Protection), hydrogen analyses, and Station Black Out (SBO). As described in EC55796, these evaluations are applicable to Cycle 19A and their respective acceptance limits are not exceeded. *As described in EC62442, these evaluations are applicable to Cycle 19B and their respective acceptance limits are not exceeded.*

Given the similarities of the physical performance of the NSF alloy with that of Zircaloy-4, introduction the NSF LUCs into the RBS core is not expected to alter in any way the predicted response of the fuel to any of the above described challenges. Previous industry experience with NSF LUCs has confirmed this expectation through several cycles of in-core exposure.

Based on the discussion above, no design basis limit for a fission product barrier as described in the USAR will be exceeded or altered.

8. Result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The *Cycle 19B* reload analyses performed by the fuel vendor utilized NRC approved methods as listed in Technical Specification 5.6.5, specifically GESTAR-II methodology, for GNF2 and GNF3 fuel described in the COLR and throughout the USAR. The GNF3 LUA and NSF LUC are introduced under the Lead Use application of the GESTAR-II methodology. Lead Use is a defined design method in GESTAR-II, thus the specific analyses required to support GNF3 LUA and NSF LUC have followed the method (GESTAR) required for RBS reloads. The GNF3 LUA and NSF LUC introduced into the RBS C19 design have met the required GESTAR-II acceptance criteria.

Therefore, Cycle 19 operation will not result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

I. OVERVIEW / SIGNATURES[1]

Facility: RBS **Evaluation # EN-2015-001 / Rev. #:** 2

Proposed Change / Document: EC64717

Description of Change: *This 50.59 (Revision 2) incorporates engineering change EC64717 for planned outage (PO-16-01) to replace identified fuel leakers. This updated loading pattern (ULP) is designated as C19C. The 50.59 Revision 1 covered EC62442 for forced outage (FO-16-01) to replace identified fuel leakers. This ULP is designated as C19B. The 50.59 Revision 0 covered EC55796 which was the original core design. The original loading pattern is designated as C19A. This 50.59 is addresses the integrated impact of C19A, C19B, and C19C core designs, since their analyses and requirements are interrelated.*

The core reload is a recurring activity for each fuel cycle. At the end of each fuel cycle, depleted fuel assemblies are discharged from the core and replaced by fresh reload assemblies. The remaining bundles resident in the core are shuffled to new locations and fresh fuel is loaded in accordance with the next cycle's core design and Reference Loading Pattern (RLP). This evaluation addresses the Cycle 19 reload changes for operation of the core including implementation of a Lead Use Assembly (LUA) program (GNF3 fuel), implementation of an Enhanced Lead Use Channel (LUC) program (NSF Channels), and for completeness the revision to the C19 RLP to replace a failed bundle. Note that the revision to the C19 RLP was performed under approved methods referenced in RBS TS 5.6.5; and the revision required no further evaluation beyond EN-LI-100 screening. As described in the preceding PAD (EC55796), this 50.59 evaluation was completed to document the impact of proposed changes to the RBS Core Operating Limits Report (COLR) which reflect C19 core operation as impacted by GNF3 LUA and NSF LUC.

EC55796 addresses changes associated with the Cycle 19 reload core into which 220 fresh GNF2 bundles and 4 fresh GNF3 LUA, fabricated by Global Nuclear Fuel (GNF), were loaded during RFO18. The GNF3 LUA use NSF channels controlled under the LUA program. In addition, 49 of the 220 fresh GNF2 reload bundles use NSF channels controlled separately under the LUC program, and 171 GNF2 reload bundles use Zircaloy-4 channels. The current COLR containing the operating limits and stability region boundaries and the reference core loading pattern (RLP) is revised to reflect operation of the Cycle 19 core.

During RFO-18 refueling, a failed (cladding failure) fuel bundle (RGE896) was removed from the core. The existence of the fuel failure had previously been identified in CR-RBS-2015-00009. This bundle was originally planned for use in the C19 RLP. Therefore, an Updated Loading Pattern (ULP) was prepared to complete the C19 core. Since the RLP configuration was used as the basis for the fuel licensing calculations, an evaluation performed on the ULP concluded that all licensing analyses remain applicable to the updated Cycle 19 core loading. This segment of the RBS design change was performed under the existing approved methods and required no further licensing evaluation; it was included here for completeness of the RBS C19 loading pattern description.

FO-16-01 was implemented to remove multiple fuel leakers from the RBS C19 core. The fuel leakers were replaced with fuel previously discharged to the Spent Fuel Pool (SFP) in RBS RF18, resulting in another Updated Loading Pattern (ULP). To distinguish between the two ULP designs, the first design from beginning of cycle (BOC) conditions evaluated in EC55796 is referred to as C19A ULP; while the second design resulting from FO-16-01 and evaluated in EC62442 is referred to as C19B ULP. This evaluation addresses the C19C ULP changes for operation of the core after removal of fuel leakers in PO-16-01. Note that the ULP revision to the C19B core was performed using approved methods referenced in the RBS TS 5.6.5. EC62442 documented that the RBS C19B ULP resulting from FO-16-01 continued to meet all required core design and licensing criteria. EC64717 documents that the RBS C19C ULP resulting from PO-16-01 continues to meet all required core design and licensing criteria. As described in the preceding PAD, this 50.59 evaluation was completed because the C19C core has been re-configured and to document the impact of proposed changes to the RBS Core Operating Limits Report (COLR) which reflect the ULP design.

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

During FO-16-01, three failed (cladding failure) fuel bundles (RGE729, SGE003, and SGE039) were removed from the core. The existence of the fuel failures had previously been identified in CR-RBS-2015-3252.Through a combination of in-core shuffles and insertion of three GNF2 bundles previously discharged to the SFP in RF18 (QGE530, QGE605, QGE607), a ULP was prepared to complete the C19B core. Each of the re-insert bundles taken from the SFP was reviewed to reload criteria from EN-NF-105, "Reload Process", including visual inspection of each to confirm no debris had been captured in the upper or lower tie plates. Note that each individual bundle replacement in the ULP configuration (shuffle and SFP re-insert) was designed to ensure that location in the C19B core would operate with a lower energy bundle than in the previous design. During PO-16-01, two failed (cladding failure) GNF2 fuel bundles (SGE123 and SGE072) were removed from the core. The existence of the fuel failures had previously been identified in CR-RBS-2016-02576.The failed bundles were both fresh reload fuel at BOC19. Through a combination of in-core shuffles and insertion of two GNF2 bundles previously discharged to the SFP in RF18 (QGE532 and QGE601, both twice burned fuel at discharge), a ULP was prepared to complete the C19C core. Each of the re-insert bundles taken from the SFP was reviewed to reload criteria from EN-NF-105, "Reload Process", including visual inspection of each to confirm no debris had been captured in the upper or lower tie plates. Note that each individual bundle replacement in the ULP configuration (shuffle and SFP re-insert) was designed to ensure that location in the C19C core would operate with a lower energy bundle than in the previous design. Since the C19B configuration was used as the basis for the fuel licensing calculations supporting the current RBS C19B COLR, an evaluation was performed on the C19C ULP; which concluded that all licensing analyses remain applicable to operation of the RBS C19C core. This segment of the RBS C19C design change was performed under the existing approved methods and demonstrated that the previous licensing evaluation remained applicable for operation of the revised core loading.

Summary of Evaluation: The RBS C19 core design has been developed using NRC approved methods, General Electric Standard Application for Reactor Fuel (GESTAR-II), which are referenced in TS 5.6.5. The C19 core will be operated in conformance with cycle specific, generic, and fuel design specific analyses provided by Global Nuclear Fuel (GNF) using the GESTAR method. The C19 core operating limits have been incorporated into the core monitoring system prior to startup of C19. LAR-2015-01 was initiated to implement the required changes to the C19 COLR.

The ULP to reconfigure the RBS *BOC19* core without the failed fuel bundle was prepared using Entergy reload procedure EN-NF-105 *(EC55796)* and by NRC approved vendor methods (GESTAR-II). Through these methods, it was demonstrated that the revised core loading continued to meet the acceptance criteria of the reload licensing analyses without the need for any changes to the established C19 core operating limits. As such, no further licensing review was required.

An Enhanced Lead Use Channel program (per GNF Document MFN 12-074, Enhanced LUC Program for NSF Channels) has been implemented in RBS C19 to introduce the use of GNF's NSF alloy for fuel channels. RBG-47552 documents the notification to NRC of the LUC program implementation at RBS. A total of 8% (49 channels) of the C19 core fuel inventory uses the NSF LUCs, all with a pre-oxidized surface finish. This is part of a continuing effort to improve the resistance to distortion of the fuel channels used at RBS. Monitoring and inspection, per the NRC SER on MFN 12-074 (see Attachments 7.012 – 7.014 of EC55158), will be implemented in RF19 and later cycles to confirm the performance of the NSF LUCs. The mechanical characteristics of the NSF LUCs (including yield and tensile strength, ductility, corrosion resistance, etc.) are very similar to those of the Zircaloy-4 channels currently in use at RBS. The NSF channel material is being introduced as a GESTAR-II Lead Use application. The NSF LUC has extensive industry experience with exposures of channels up to 3 & 4 two-year cycles at several BWR units (including BWR/6). Monitoring and inspection results for these applications have demonstrated favorable performance.

A LUA program has been implemented in RBS C19 to introduce use of the GNF3 fuel assembly. RBG-47551 documents the notification to NRC of the LUA program implementation at RBS. A total of 4 GNF3 LUA were included in the RBS C19 core. The LUA is a part of a continuing industry effort to expand the operating experience with new fuel designs (improved performance in fuel efficiency). Monitoring and inspection of the GNF3 performance will be implemented in RF19 and later cycles. Under its LUA classification in the GESTAR-II method, the GNF3 fuel has been designed such that its mechanical and nuclear characteristics were confirmed to meet the required design acceptance criteria. GNF3 design characteristics are very similar to those of the

GNF2 design already in operation at RBS. The GNF3 fuel will be operated in conformance with cycle specific, generic, and fuel design specific analyses provided by Global Nuclear Fuel (GNF) using the GESTAR method. The RBS C19 core was designed such that the GNF3 will operate with margin to the specified operating limits, and such that the GNF3 LUA will not be limiting at any time in C19. The GNF3 specific C19 core operating limits are incorporated into the core monitoring system prior to startup of C19. These fuel specific operating limits are addressed in LAR-2015-01 as a part of the changes to the C19 COLR.

The RBS C19C ULP core design (PO-16-01) has been developed using NRC approved methods, General Electric Standard Application for Reactor Fuel (GESTAR-II), which is referenced in TS 5.6.5. The C19C ULP core will be operated in conformance with cycle specific, generic, and fuel design specific analyses provided by Global Nuclear Fuel (GNF) using the GESTAR method. EC64717 documents that the C19A core operating limits continue to meet all core design and licensing criteria, and thus no changes are required for the current COLR operating limits. The only COLR change required is an update to one of its references; which cites the vendor evaluation of the ULP. The change in core inventory will be incorporated into the core monitoring system prior to startup of the C19 ULP (C19C). LBDCR-2016-01, Rev.1 was initiated to implement the required changes to the C19C COLR.

The C19A ULP reload core design included 220 fresh GNF2 bundles and 4 fresh GNF3 LUA, fabricated by Global Nuclear Fuel (GNF), were loaded during RFO18. SE #2015-001, Rev.0, OSRC Meeting 15-004 documented the introduction of the GNF3 Lead Use Assemblies (LUA) and of the GNF NSF Enhanced Lead Use Channels (LUC). The RBS C19C ULP does not impact the LUA or LUC components or programs as described in SE #2015-001, Rev.0, OSRC Meeting 15-004; thus the C19A ULP SE continues to apply to operation of the RBS C19C ULP. The C19C ULP to reconfigure the core without the failed fuel bundles was prepared using Entergy reload procedure EN-NF-105 and by NRC approved vendor methods (GESTAR-II). Through these methods, it was demonstrated that the revised core loading continued to meet the acceptance criteria of the reload licensing analyses without the need for any changes to the established C19 core operating limits. The current COLR was revised to reflect operation of the C19C ULP core. Note that the only change concerns an update to COLR Reference 3.1.5.

The reload licensing basis for the RBS C19 core is described in Ref.3.1.1 of the RBS C19 COLR, with evaluations of deviations in core design provided in COLR Ref.3.1.5. The latest update to COLR Ref.3.1.5 documents the fuel vendor reload licensing evaluation of the changes made with the RBS C19C ULP. The procedure for evaluating any core design deviations is part of the NRC approved reload licensing methodology, GESTAR-II, Section 3.4.2, "Acceptable Deviation from Reference Core Design". If any of the acceptance criteria in Section 3.4.2 are exceeded, the vendor must re-examine all the affected licensing parameters to ensure there is no adverse impact per GESTAR-II Section 3.4.3, "Re-examination Bases". The C19C core design met all of the Section 3.4.2 criteria with the exception of 'Locations of Fresh Reload Bundles'. Because the two failed bundles were fresh reload assemblies from interior core locations and were replaced with other than fresh bundles,C19C required a full re-examination of the licensing parameters supporting the operating limits reported in the COLR Ref.3.1.1.The vendor evaluation in COLR Ref.3.1.5 includes re-examination of the following 8 parameters.

1,2) Scram Reactivity Insertion and Dynamic Void Coefficient

For Pressurization transients, the scram reactivity insertion was not degraded relative to the licensing basis for EOC & MOC conditions in the C19C ULP core configuration. The dynamic void coefficient was confirmed to be within the range analyzed for the licensing basis of EOC & MOC conditions. Therefore, the pressurization event Operating Limit Minimum Critical Power Ratio (OLMCPR) results for the core-wide Anticipated Operational Occurrence (AOO) reported in the COLR remain valid for the C19C ULP core.

In the RBS C19A design, all Thermal Over-Power (TOP) and Mechanical Over-Power (MOP) results for the Loss of Feedwater Heating (LFWH) event were within design limits and the LFWH event did not set the required OLMCPR. Due to small margin in the LFWH TOP results (with respect to setting the OLMCPR), the vendor ran confirmatory evaluations for the C19C ULP core configuration. The results indicated that the previous TOP/MOP results for the LFWH event remain valid for the C19C ULP.

3) Peak Fuel Enthalpy during Rod Drop Accident

Per the GESTAR-II methods, since RBS is a 'banked position withdrawal sequence plant', the control rod drop accident analysis is not required.

4) Cold Shutdown Margin

The predicted shutdown margin for the C19C ULP at PO-16-01 exposure (8924.6 MWd/ST) is greater than BOC19 (2.8%), which meets the design limit minimum value of 1%. The minimum shutdown margin for the cycle occurred at BOC19. A shutdown margin surveillance test (STP-050-3601, "Shutdown Margin Demonstration") is conducted at startup (at BOC19A and startup of C19B & C19C) to demonstrate compliance with technical specifications.

5) Standby Liquid Control System (SLCS) Shutdown Margin

The minimum SLCS shutdown margin was calculated at 3% in the C19A design; which meets the design limit minimum value of 1%. This is a beginning of cycle value that increases throughout the cycle. As described earlier in EC62442 and SE# EN-2015-001, Rev.1, the overall reactivity of the C19B core was reduced from that of the C19A core design. Since the overall reactivity of the C19C ULP core is reduced from that of the C19B design, the previously determined C19A minimum shutdown margin remains appropriate for the C19C ULP design.

6) Change in Critical Power Ratio due to a Mislocated Fuel Assembly

RBS meets the requirements (described in GESTAR-II) for classifying the fuel loading events (mislocated and misoriented) as an Infrequent Incident; therefore cycle specific loading event analyses are not required.

7) Rod Block Monitor Response to a Rod Withdrawal Error (RWE)

The C19A design demonstrated that all TOP/MOP results for the RWE event were within design limits and did not set the required OLMCPR. Due to small margin in the MOP results for the RWE (with respect to setting the OLMCPR), the vendor ran confirmatory evaluations for the C19C ULP core configuration. The results indicated that the previous TOP/MOP results for the RWE event remain valid for the C19C ULP.

8) Safety Limit MCPR (SLMCPR)

The Safety Limit analysis assumes core quadrant fuel bundle symmetry. The introduction of the two second burn bundles into the interior locations designed for fresh reload fuel in the C19A configuration created asymmetries. Exposure and reactivity asymmetries tend to improve the SLMCPR since they increase radial peaking. Increased radial peaking reduces the total number of rods susceptible to boiling transition, thus reducing the SLMCPR value. Therefore, the SLMCPR confirmation performed for the C19A design remains applicable for the C19B ULP.

In addition to these evaluations, the vendor also performed analysis of the limiting reload validation matrix (RVM) cases that are used to confirm the Enhanced 1A (E1A) stability solution installed at RBS. The limiting RVM case 6 was evaluated and found to meet the required acceptance criteria using the C19C ULP core configuration.

Based on these evaluations, using NRC approved methods, the vendor demonstrated that the results previously reported in COLR Ref.3.1.1 remain valid for operation of the C19C ULP core.

Background Information

Mechanical

The GNF2 fuel mechanical design has been reviewed for use at River Bend (50.59 Evaluation 2011-004, OSRC

Mtg# 11-006). No unusual failure modes or increased failure frequency have been identified for this fuel design. This is the 3rd cycle of operation for GNF2 fuel and this fuel design has accumulated operational experience at RBS and other plants (including Entergy BWR units) with no significant problems. The bundles will operate within the power history assumptions in the fuel mechanical analyses and will experience exposures within the analyzed burnup limits of the mechanical designs, including those bundles that will be irradiated for a third cycle. Mechanical design analyses [USAR 4.2.1] have been performed with NRC-approved methodology to evaluate mechanical criteria including cladding steady-state strain and stresses, transient strain and stresses, fatigue damage, creep collapse, corrosion, hydrogen absorption, fuel rod internal pressure, etc. All parameters were found to meet their respective design limits for the Cycle 19 core. Although an increased channel bow condition can result in increased friction between the control blade and its corresponding fuel assemblies, control rod settle and insertion testing (OSP-0061) will continue to be performed during Cycle 19 per the guidance provided in fleet procedure EN-RE-216 to ensure that any increased axial friction loads on the channel and fuel assembly load chain remain below acceptable limits. Numerous previously irradiated C19 fuel assemblies (total of 31) were re-channeled with new Zircaloy-4 channels (identical replacements) to enhance the C19 core in terms of its resistance to channel distortion (i.e., reduce potential for channel bow).

The ULP to reconfigure the RBS C19 core without the failed fuel bundle was prepared using Entergy reload procedure EN-NF-105 and by NRC approved vendor methods (GESTAR-II). Through these methods, it was demonstrated that the revised core loading continued to meet the fuel mechanical acceptance criteria of the reload licensing analyses without the need for any changes to the established C19 core operating limits.

The mechanical characteristics of the NSF LUCs (including yield and tensile strength, ductility, corrosion resistance, etc.) are very similar to those of the Zircaloy-4 channels currently in use at RBS. Use of NSF LUC has been previously addressed for operation in the GGNS C20 core and found acceptable (SE-2014-001). The mechanical performance of NSF channels for GNF2 fuel is bound by that of channel designs previously used at RBS, such as Zircaloy-2 and Zircaloy-4.

The GNF3 mechanical design has been reviewed for use at RBS. No unusual failure modes or increased failure frequency have been identified for this fuel design. The bundles will operate within the power history assumptions in the fuel mechanical analyses and will result in exposures within the analyzed burnup limits of current fuel mechanical designs. The mechanical compatibility of the GNF3 LUA with co-resident GNF2 fuel in the RBS core has been reviewed and found acceptable in EC53147 & EC54888 (fuel receipt), in EC55250 (fuel shuffle), and EC55796 (C19 Reload). The GNF3 channels are fabricated using the NSF alloy, but are controlled under the LUA program and exclusive from the NSF LUC program. The GNF3 bundle and channel design was reviewed in EC55796 and found acceptable for operation.

The GNF3 bundles were designed using approved vendor methods for LUA design in GESTAR-II (referenced in RBS TS 5.6.5), and the methods are capable of analyzing all aspects of the new fuel design. The GNF3 LUA have been analyzed for RBS C19 specific operations such that they will not be the most limiting fuel assemblies at any time during C19 operation with respect to compliance with fuel mechanical acceptance criteria. Such analyses shall be performed for each subsequent cycle of GNF LUA operation in the RBS core to confirm continued operations in this manner.

Nuclear

The neutronic characteristics of the Cycle 19 core design have been considered in the safety analysis. The nuclear characteristics of the reload GNF2 fuel remain unchanged from previous RBS reloads. Since the ULP design changes were performed in accordance with approved methods and only involve previously irradiated GNF2 fuel, there will be no impacts on the nuclear response of the core. The nuclear characteristics of the NSF LUCs are very similar to that of currently used Zircaloy-4 fuel channels, no nuclear related impacts are expected from introduction of the NSF LUCs. The nuclear characteristics of the GNF3 fuel are similar to those of the GNF2 bundles they displaced in the RBS C19 reload fuel. Their placement in the core was analyzed to ensure they will not be the most limiting fuel assemblies in the core at any time during operation of C19. Such analyses shall be performed for each subsequent cycle of GNF LUA operation in the RBS core to confirm continued operations in this manner. Adequate shutdown margin has been predicted by analysis and will be confirmed during startup tests per Technical Specification 3.1.1. In addition, the hold-down capability of the standby liquid control system has been confirmed. Therefore,

the probability of inadvertent criticality has not been increased by the introduction of GNF3 LUA and NSF LUC in the Cycle 19 reload fuel.

Channel General and Safety Related Design Functions

The fuel channel component includes general design functions and safety related design functions.

The general design functions include:

1. The channel forms the flow path shell for fuel bundle coolant flow.
2. The channel provides surfaces for control rod guidance in the reactor core.
3. The channel provides structural stiffness to the fuel bundle during lateral loadings applied from fuel rods through the fuel spacers
4. The channel forms the coolant flow leakage path at the channel/lower tie plate interface.
5. The channel transmits fuel assembly seismic loadings to the top guide and fuel support of the core internal structure.

The safety functions of the fuel channels include:

1. The four channels in a cell establish the pathway through which the control blade moves. The lateral stiffness of the channel prevents channel buckling and ensures that the pathway remains available during design basis events, such as an earthquake. Excessive bulge and bow of channels may affect the movement of rods and the scram time. Therefore, the fuel channels impact the capability to shut down the reactor and maintain it in a safe shutdown condition.
2. The channel provides a barrier to allow parallel coolant flow paths and provides a heat sink which cools the outer row of fuel rods during a LOCA event.
3. The channel provides a barrier to fuel rod failure propagation from one fuel assembly to others by maintaining separation of fuel rods and by restricting the movement of debris that may be associated with the initial failure. Therefore, the fuel channel may help to mitigate the consequences of certain severe accidents.

The NSF alloy has demonstrated improved bow characteristics in prior lead use programs. Hence, the operational experience in these lead use programs suggests that NSF will perform with greater resistance to the various forms of channel distortion (i.e., fluence gradient-induced bow, shadow corrosion-induced bow) and have corrosion performance similar to the current Zircaloy-4 channels in the RBS core. The previous LUC programs provide a degree of confidence that there will be no unanticipated issues in the behavior and performance as compared to Zircaloy. The GNF3 NSF channel design was reviewed and found acceptable for operation in the RBS core in EC55796.

The material properties of the NSF alloy are sufficiently similar to current channel materials such that NSF channels are expected to be dimensionally, structurally and mechanically compatible with Zircaloy-4 channels within a four bundle cell. Differences in the mechanical, thermal, and nuclear properties of NSF, as compared to Zircaloy-4, are minor and do not affect the methodology used to design and analyze the channels. Similarly, the minor alloy differences between NSF and Zircaloy have little to no significance to core design, transient, stability or ECCS-LOCA analyses.

Is the validity of this Evaluation dependent on any other change? ☐ Yes ☒ No

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ Yes ☒ No

Preparer: Jim Head / See AS* / ESI / Fuels Analysis / 6/21/2016
Name (print) / Signature / Company / Department / Date

Reviewer: W. Steelman / See AS* / EOI / Licensing / 6/22/2016
Name (print) / Signature / Company / Department / Date

OSRC: Marvin Chase / 6/22/2016
Chairman's Name (print) / Signature / Date

OSRC-RBS-2016-009
OSRC Meeting #

* EC 0000064717

II. 50.59 EVALUATION

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. **Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR?** ☐ Yes ☒ No

BASIS:

RBS Cycle 19 operation will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR. The precursors to these events are independent of the core design and the frequency classifications reported in USAR Chapter 15 are unaffected by the core parameters. The following considerations support this conclusion.

Core Operating Limits

Operating Thermal Limits

Having followed approved design methods for the *RBS C19C core design*, power operation within the core operating limits (Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR), Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)) will ensure that the appropriate safety criteria are met during normal operations, Anticipated Operational Occurrences (AOO), or accidents. The precursors to these events are independent of the core design and the frequency classifications reported in SAR Chapter 15 are unaffected by the core parameters. Therefore, the core parameters in the COLR have no effect on the probability of the occurrence of any accident described in the SAR.

APRM Limits and Stability Region Boundaries

As with the normal reload review process for any core design, the stability performance of the reload design must be evaluated against various acceptance criteria specified in the E1A stability solution. Analyses have been performed to demonstrate that the revised E1A region boundaries for Cycle 19 met the channel and core decay ratios acceptance criteria. The change in E1A stability region boundaries implemented in the RBS C17 reload design will continue to avoid a thermal-hydraulic instability during the Cycle 19 operation. The *RBS C19C ULP core design* with NSF LUC and GNF3 LUA has been analyzed and validated to meet all required stability performance criteria. Therefore, the probability of a thermal hydraulic instability has not increased.

Design Basis Accidents (Limiting Faults)

The design basis accidents analyzed in support of Cycle 19 are the control rod drop accident (CRDA), the fuel handling accident (FHA), and the loss of coolant accident (LOCA). The FHA is discussed separately.

The probability of the occurrence of design basis accidents is not dependent on the core configuration. No changes to the plant design are required for the *Cycle 19C core*. The *Cycle 19C core loading* will not affect the precursors to any of the Chapter 15 events, including LOCA analyses.

The Control Rod Drop Accident (CRDA) results from a failure of the control rod-to-drive mechanism coupling after the control rod becomes stuck in its fully inserted position. Although a channel bow condition can result in friction between the control blade and its corresponding fuel assemblies, analyses have shown that there would not be sufficient friction to result in a mechanical failure of the coupling. Additionally, the control rod drive mechanism would not produce enough force to result in a mechanical failure of the coupling even if the channel bow was so severe that the assemblies would preclude blade movement. Channel bow associated with GNF2 reload fuel at high exposure is no more than the channel bow associated with previous RBS reload fuels (i.e., ATRIUM-10, GE14). The addition of NSF LUCs and the GNF3 LUA is not expected to have any negative impact on channel performance in the core; based on industry experience with other LUC programs with multiple cycles of exposure, the performance of the NSF LUCs has been similar to or improved over that of the Zircaloy-4 channels already present in the RBS core. As such, channel bow is not considered a precursor to the CRDA, and any increased bow associated with the high exposure fuel bundles would not increase the probability of this event.

Fuel Handling Accident and Fuel Loading Error

The FHA related impacts of GNF2 were previously evaluated for RBS application in RBS C17. *Since the RBS C19C ULP makes use of re-insert GNF2 fuel from the same RBS C17 reload batch fuel, the FHA related impacts from use of GNF2 continue to apply.* As described in RBS C19 Fuel Receipt EC54888, the GNF3 design is very similar in its overall physical characteristics to that of GNF2 and that its interface with handling equipment is likewise similar to GNF2, there is no expectation for an increase in the probability of occurrence of a FHA.

The fuel loading error (FLE) event involves improper loading of a fuel bundle into the core, related to human error associated with the ability to properly identify a bundle by its serial number identification and ability to determine the proper orientation of the bundle when loading into the core. These factors are driven by design features of the upper tie plate of the bundle, where the bundle ID is machined and orientation queues are built into the UTP design. The identification and orientation features of the GNF3 design are consistent with that of GNF2. Thus, there is no increase in the probability of occurrence of an FLE event.

On these bases, the probability of occurrence of accidents previously identified in the SAR is not increased for the Cycle 19C core.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR? ☐ Yes ☒ No

BASIS:

No plant modifications are required to accommodate the Cycle 19 core design. Related specifically to the fuel in the reloaded core, the only additional loads placed on plant equipment would be due to changes in the characteristics of the fuel as it interfaces with other core components and functions; such as stability, decay heat, reactor internal pressure differences, reactor internal structural impacts, transient responses, neutron fluence, radiation source term, recirculation system impacts, changes to EOPs, fuel storage criticality, or other severe accident related affects. All of these topics were specifically addressed by the fuel vendor's assessment for introduction of the GNF3 LUA design into the RBS core. All areas of review found acceptable results; with either no impact identified or minor changes that are within the current RBS design and licensing basis limits for fuel and core design (i.e., GNF3 accounted for in Thermal Limits results applied in the C19 COLR, GNF3 design part of the bases for revised RBS SFP Criticality analysis, etc.). The C19 reload evaluation has shown that the GNF3 LUA are compatible with co-resident fuel and core components in terms of its form, fit, and function. GNF3 does not introduce an increased potential for interference with core components (such as control rod blades, instrument tubes, etc.). Therefore, GNF3 introduction into the RBS C19 core does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of core components.

The only additional loads placed on plant equipment associated with the reload GNF2 fuel would be due to increased friction between the control blades and excessively bowed channels. As described in EC55796, the ULP design includes fuel channel designs previously used at RBS, and the re-channel of numerous previously irradiated fuel bundles in the RBS C19 core will reduce the potential for occurrence of excessive channel bow. *As described in EC64717, the C19C ULP design likewise continues to use previously proven channel designs. The RBS C19C ULP design has been analyzed for the potential to develop excessive channel bow and found acceptable.* Introduction of NSF LUCs does not impact previously evaluated SSC's or their malfunction as described in the USAR since its properties are very similar to those of the co-resident Zircaloy-4 fuel channels in the RBS core. Based on previous experience with bowed fuel at RBS and other BWR-6 units, increased control blade friction can result in increased control rod settle times but is not expected to significantly impact scram times. Technical Specification scram time testing and control rod settle and insertion testing will continue to be performed during Cycle 19, in accordance with guidance provided in fleet procedure EN-RE-216. These actions would identify any potential scram time or other impacts such that appropriate corrective actions are taken. These actions will ensure that any increased control blade friction loads are not sufficient to cause any failures associated with the control blades or the control blade drive system, the fuel assembly load chain, or the vessel internals. Regarding NSF LUC conditions and requirements specified by NRC in their SER of the Enhanced LUC program, additional monitoring (settle testing and scram time testing) and inspections must be performed to confirm the performance of the LUCs. This additional monitoring and inspection will begin in RFO19 and for operation of the NSF LUC in RBS Cycle 20 and beyond.

Analysis of Reactor Internal Pressure Differences (RIPDs) resulting from the introduction of a new fuel type with different hydraulic characteristics is done to ensure design basis criteria are met. The pressure across the RBS C19 core is the result primarily of the dominant fuel in the core, GNF2.The introduction of four GNF3 LUA in the 624 bundle RBS core has very little impact on the overall core hydraulic characteristics. In addition, the GNF3 hydraulic performance is bounded by past fuel types used at RBS. *The changes introduced with the RBS C19C ULP design continue to have the same hydraulic performance characteristics as found in the C19A ULP design (i.e., each core location has the same fuel design type in both the C19A and C19C ULP cores).*

A reactor internal structural assessment has been performed for the reload fuel. This assessment included normal, upset, emergency and faulted condition loads on the reactor internals and concluded that the reactor internals remain qualified for the reload fuel. The evaluation of GNF3 LUA found no adverse impacts to the structural integrity of the reactor internals with regard to RIPD or seismic loading. The GNF3 assemblies have an insignificantly greater weight than GNF2. The loading plan for the GNF3 LUA places one bundle in each of the four quadrants of the RBS core, thus

avoiding any local concentration of the added weight. When analyzed on a cell basis, the additional weight averaged over a 4-bundle cell with 3 co-resident GNF2 assemblies is approximately 0.5%, which is considered negligible relative to structural integrity. The magnitude of the increase in bundle weight of GNF3 versus GNF2 is comparable to past RBS reloads of mixed fuel types such as GE14 versus ATRIUM-10. Based on evaluating the combined effects of RIPD, seismic, and fuel assembly weight, the introduction of four GNF3 LUA will have an insignificant effect on the structural integrity of the reactor internal components.

The dominant fuel type, GNF2, dictates the seismic behavior if the RBS fuel core response. Introduction of the four GNF3 LUA does not have any significant impact on the mass and stiffness of the RBS C19 fuel core, and does not change the fuel fundamental frequency. The GNF3 fuel lift margin and control rod guide tube lift forces are bounded by previous fuel types used at RBS. Thus the design basis fuel lift loads remain valid for the four GNF3 LUA. The seismic capability of the GNF3 spacers was confirmed through seismic load testing. *The changes introduced with the RBS C19C ULP maintain the same ratio of GNF2 to GNF3 inventory as the C19A design.* The current design seismic qualification of the RBS C19 core in supporting operation with four GNF3 LUA remains unchanged

A conservative vessel over-pressurization analysis has been performed, which shows that the vessel pressure limit is not exceeded. The results of the analysis were reported in the Supplemental Reload Licensing Report (SRLR) for the RBS C19 core, in which the GNF3 LUA was explicitly modeled. The RBS C19 core with GNF3 LUA was shown to continue to meet the required ASME overpressure protection criteria. *The evaluation performed in support of EC64717 found that the RBS C19 SRLR results remain unchanged by the RBS C19C ULP design.*

The impact on the vessel fluence associated with the RBS C19 reload was evaluated in EC55796. It was found that the current RBS vessel fluence analysis remains valid with the C19 reload core design. Given the interior location of the GNF3 LUA in the RBS C19 core with respect to the vessel wall, GNF3 does not impact vessel fluence in operation of the C19 core and the current RBS P/T curve remains applicable. Thus GNF3 does not result in an increase in the likelihood of occurrence of fluence induced damage to the vessel. *The evaluation performed in EC64717 found that the RBS C19C ULP design does not impact the vessel fluence; the current RBS vessel fluence analysis remains valid with the C19C reload core design.*

The containment response is determined primarily by fuel-independent parameters such as RHR heat exchanger performance, ECCS coolant injection flow, service water temperature, and initial suppression pool temperature. As previously described, introduction of the GNF3 LUA does not result in more than a minimal impact on the containment response.

With regards to post-LOCA hydrogen control, per 10CFR50.44 RBS must have a system that will mitigate hydrogen from a metal-water reaction involving 75% of the core cladding surrounding the active fuel region. The containment igniter system meets this requirement for the current GNF2 reload core. The addition of GNF3 LUA results in a bundle specific comparison of approximately 3% increased cladding surface in the active fuel region (i.e., GNF3 has approximately 3% more cladding surface in the active fuel region than GNF2). However, comparison of the total active surface area in the C19 core configuration shows that it is bounded by that of the C18 configuration. *The ratio of GNF2 to GNF3 assemblies in the RBS C19C core is identical to that of the RBS C19A design.* Therefore, introduction of the GNF3 LUA does not have more than a minimal impact post-LOCA hydrogen control.

The precursors to any malfunction of equipment important to safety are not affected by the Cycle 19C reload core. Therefore, there is not more than a minimal increase in the likelihood of an occurrence of a malfunction of a SSC important to safety previously evaluated in the USAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR? ☐ Yes ☒ No

BASIS:

The acceptance criteria reported in USAR Section 15.0.3.1 and the Technical Specifications are satisfied for each event classification. Core operating limits have been developed to ensure that moderate frequency events do not violate the MCPR safety limit or fuel cladding strain limits. The consequences of infrequent events have been shown to meet the appropriate acceptance criteria while the individual acceptance criteria for the limiting faults have been demonstrated to be satisfied. Satisfying these acceptance criteria ensures an upper bound on the potential release of radiological source term, thus limiting the resulting consequences of the event. As such, the consequences of infrequent events and limiting faults described in the USAR are unchanged for the Cycle 19 reload core. The following considerations support these conclusions.

Anticipated Operational Occurrences (AOO) Events

The Cycle 19 core operating limits have been developed with NRC-approved methodologies such that the MCPR safety limit and the fuel cladding strain limit will not be violated by any analyzed moderate frequency transient initiated from any statepoint available to RBS. *The evaluation performed in support of EC64717 found that the RBS C19 core operating limits (including stability region boundaries) remain unchanged by the RBS C19C ULP design. Fuel vendor analysis of the C19C core design included confirmatory evaluations of the Loss of Feedwater Heating (LFWH) and Rod Withdrawal Error (RWE) events. In addition, the limiting reload validation cases were evaluated for the RBS E1A stability regions. In every evaluation, the C19C results remained within design limits, confirming that the existing licensing conclusions presented in RBS COLR Ref.3.1.1 remain valid for operation of the C19C core.* As such, no fuel failures are expected to result from any moderate frequency event. Thus the radiological source term and the potential extent of its release from the fuel have not been increased with the introduction of GNF3 LUA in the RBS C19 core. These analyses considered RBS-specific operational modes such as Maximum Extended Operating Domain (MEOD), Single Loop Operation (SLO), Feedwater Heater(s) out of Service (FHOOS), Final Feedwater Temperature Reduction (FFTR), Pressure Regulator Out Of Service (PROOS), Main Turbine Bypass Out Of Service (MTBOOS) and End Of Cycle –Recirculation Pump Trip (EOC-RPT) inoperable. These core operating limits consist of MCPR and LHGR curves that are functions of flow, power, and exposure for both GNF2 and GNF3 fuel. These core operating limits have been incorporated into the core monitoring system. These limits consider conservative channel bow assumptions that bound the current measured bow data and the expected increased bow associated with the highly exposed fuel. Introduction of the NSF LUC is not expected to impact the core with respect to performance of the fuel channels. Industry experience with other applications of NSF LUC has yielded favorable results. GNF3 LUA have been loaded into the RBS C19 core such that they will operate with margin to the thermal limit requirements. Approved methods were used to model the GNF3 LUA and confirm they respond in a similar manner to the GNF2 assemblies they replace in the RBS C19 core. Thus introduction of the GN3 LUA and NSF LUC to the RBS C19 core design does not result in an increase in the consequences any AOO event.

Infrequent Events

The fuel loading error (*i.e.,* misoriented and mislocated) is considered an infrequent event. The consequence of this event has been evaluated in accordance with the GESTAR requirements and shown to meet the respective acceptance criteria. Radiological analyses using the alternative source term (AST) have been performed to ensure that these events will not result in an increase in offsite or control room dose or doses greater than their respective acceptance criteria. Evaluation of the *RBS C19C core design in EC64717*, including GNF3 LUA and NSF LUC demonstrated that the current radiological source term for RBS remains bounding. The radiological analyses reflect operation as described in the USAR.

Limiting Faults

The limiting faults at RBS include the fuel handling accident, the control rod drop accident, and the design basis LOCA. The radiological analyses for these events have been developed as part of the AST effort assuming certain core parameter values such as radial peaking factor and fuel burnup. Core parameters assumed in the dose analysis were reviewed and found to be bounding for the Cycle 19 core. The number of rod failures in a CRDA or FHA using GNF2 fuel remains bounded by the assumption used in the dose analyses. The Cycle 19 core (with inclusion of the GNF3 LUA) parameters have been shown to be bounded by the RBS AST source term and radiological analyses. For the Loss Of Coolant Accident (LOCA), MAPLHGR operating limits and single-loop multipliers have been developed for the Cycle 19 core configuration, including GNF3 LUA, such that the requirements of 10CFR50.46 are satisfied. The containment response for the Cycle 19 core, including GNF3 LUA, was found to not result in more than a minimal impact on previous GNF2 design based containment system response analysis. Review of the potential degraded core hydrogen generation has shown that the RBS C18 core configuration remains bounding with respect to the Cycle 19 core configuration, including GNF3 LUA. *The evaluation performed in EC64717 found that the RBS C19C ULP design does not impact the current AST or radiological analyses; the current RBS source term and dose analyses remain valid with the C19C reload core design. The ratio of GNF2 to GNF3 assemblies in the RBS C19C core is identical to that of the RBS C19A design.* Review of the Cycle 19 core, including GNF3 LUA, has shown that there will not be more than a minimal impact on the seismic/LOCA response. The GNF3 LUA design has no effect on the LOCA and Transient Analyses and the consequence from these events is not impacted.

Compliance with licensing acceptance criteria for the CRDA is assured through adherence to the Banked Position Withdrawal Sequence (BPWS) where analyses have generically demonstrated large margin to licensing limits on enthalpy insertions. Given the similarities of the nuclear characteristics of the GNF3 and GNF2, the generic analysis remains applicable to RBS operation with GNF3 LUA.

With respect to LOCA performance of the RBS C19 core, the GNF3 bundles will be restricted to a peak linear heat generation rate (PLHGR) equivalent to that of GNF2. The GNF3 licensing basis peak clad temperature (LBPCT) was determined using approved methods and demonstrated conservative margin to the 10CFR50.46 PCT limit. As previously discussed, the change in post-LOCA hydrogen produced from a metal-water reaction with the GNF3 fuel in the C19 core configuration remains bounded by that from the C18 configuration. Since the LBPCT and maximum local oxidation remain within licensing limits, a coolable geometry is assured. Likewise, introduction of GNF3 does not impact the reflooding capability of the ECCS or operation of the core spray systems, thus also does not impact long term cooling. Therefore, the acceptance criteria of 10CFR50.46 remain satisfied with introduction of the GNF3 LUA.

The FHA was analyzed previously for introduction of GNF2 in RBS C17. *Since the RBS C19C ULP makes use of re-insert GNF2 fuel from the same RBS C17 reload batch fuel, the FHA related impacts from use of GNF2 continue to apply.* The analysis was performed for GNF3 bundles in the RBS core design. Given that the GNF3 LUA are loaded in non-limiting locations of the C19 core, they will accumulate less bundle exposure and less bundle source activity. Despite this loading restriction, the analysis applied the same conservative assumptions regarding exposure and radial peaking factor, which increased the assumed bundle source activity more than that associated with the slight increase in fuel mass of GNF3 versus GNF2 bundles. The results of the FHA analysis demonstrated that even given these conservative assumptions, the consequence multiplier of the GNF3 fuel remains less than unity and is therefore bounded by the consequence of a FHA involving GNF2 fuel, confirming that the GNF3 LUA have no impact on the dose consequences of the FHA.

The core radiation source term impact from the introduction of GNF3 is inconsequential since its design is similar to that of GNF2. As previously described, the C19 reload parameters affecting the radiological source term were compared against the acceptance criteria and found to remain bounded. Therefore, GNF3 LUA in the C19 core are expected to have no significant impact on the RBS core radiological source term. Radionuclide concentrations in reactor water and steam are not dependent on fuel type but on operating conditions of the core, and since GNF3 will not affect core performance; it will not affect coolant radiation sources. Therefore, GNF3 will not affect the dose consequences of postulated reactor coolant system accidents such as Main Steam Line Break.

A recirculation pump seizure event while in single loop operation is also a limiting fault, but is conservatively evaluated against the AOO acceptance criteria. As the AOO acceptance criteria (MCPR) are satisfied, the limiting fault acceptance criteria are satisfied. This was demonstrated for the introduction of GNF2 in RBS C17. *Since the RBS C19C ULP makes use of re-insert GNF2 fuel from the same RBS C17 reload batch fuel and continues to meet the AOO acceptance criteria (using the same MCPR limits as the C19A design), the limiting fault acceptance criteria are satisfied.* The introduction of four GNF3 LUA into RBS C19 will not affect the results of the analysis due to the large conservatisms assumed in the calculation of the void coefficient. For the GNF3 LUA, the same critical power correlation as used for GNF2 (GEXL17) has been conservatively adjusted for application to GNF3. Therefore, the previously determined GNF2 SLO pump seizure limits apply to GNF3.

Based on the review above, it is concluded that the Cycle 19C reload design will not increase the consequences of an accident previously evaluated in the USAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR? ☐ **Yes** ☒ **No**

BASIS:

All equipment important to safety will function in the same manner with the *Cycle 19C ULP core* as with the previous core design. The proposed reload does not require new hardware or modification of existing hardware. Any malfunction of key plant components as described in USAR which could impact fuel integrity has been factored into the Cycle 19 transient and accident analyses. The consequences of these malfunctions have been shown to meet their respective acceptance criteria and to remain unchanged and bounded for Cycle 19 operation. Specific to the Cycle 19 reload design with the introduction of GNF3 LUA, it has been shown to be compatible with the other co-resident GNF2 fuel in the core. GNF2 fuel was introduced at RBS in C17, and it has demonstrated acceptable performance in reload evaluations and operations for C17 and C18. As previously described, the performance of GNF3 as determined using approved methods is predicted to be similar to that of GNF2. Introduction of the NSF LUC is compatible with the Zircaloy-4 channels currently used in the RBS core, and is expected to provide comparable or improved performance during operation. Specific topics are discussed below:

Core Operating Limits

The SLMCPR and AOO analyses are based upon Cycle 19 core design to determine operating thermal limits, using the NRC approved methodology. Operation within these limits during normal steady state operation provides assurance that MCPR safety limits and fuel thermal/mechanical design limit are not exceeded due to the most severe AOO event. An AOO event assumes a single equipment failure such as load reject or turbine trip which causes a spike in reactor power and heat flux. The transient analyses have included such severe events so that safety limits are always maintained. The applicability of the current stability region boundaries depicted in the COLR has been confirmed for Cycle 19. *The evaluation performed in support of EC64717 found that the RBS C19A core operating limits (including stability region boundaries) remain unchanged by the RBS C19C ULP design. Fuel vendor analysis of the C19C core design included confirmatory evaluations of the Loss of Feedwater Heating (LFWH) and Rod Withdrawal Error (RWE) events. In addition, the limiting reload validation cases were evaluated for the RBS E1A stability regions. In every evaluation, the C19C results remained within design limits, confirming that the existing licensing conclusions presented in RBS COLR Ref.3.1.1 remain valid for operation of the C19C core.* As such, no fuel failures are expected to result from any AOO event. The loading the GNF3 LUA has been designed such that they will not be the limiting fuel in the core at any time in C19 operations. The core loading for the GNF3 LUA has been designed such that the GNF3 will operate with margin to the operating limits. Thus the RBS C19 core is designed such that the potential release of bundle source radionuclides remains bounded by the limiting source term analysis. Therefore, the consequences of a malfunction of a safety-related structure, system or component have not increased.

Design Basis Accident

The design basis accidents reviewed in support of Cycle 19 are the control rod drop accident (CRDA), the fuel handling accident (FHA), and the loss of coolant accident (LOCA). The core parameters assumed in the RBS dose analyses remain bounding with respect to the Cycle 19 operating core parameters with introduction of GNF3 LUA. *The evaluation performed in EC64717 found that the RBS C19C ULP design does not impact the current AST or radiological analyses; the current RBS source term and dose analyses remain valid with the C19C reload core design.*

Therefore, Cycle 19C operation will not increase the consequence of a malfunction of equipment important to safety evaluated in the SAR.

5. **Create a possibility for an accident of a different type than any previously evaluated in the USAR?** ☐ Yes ☒ No

BASIS:

The Cycle 19 GNF3 fuel has been shown compatible with GNF2 fuel that was inserted in previous cycles. There are no changes in the operation of any system; highly exposed fuel bundles are replaced with fresh bundles. The malfunction of key components as described in USAR are analyzed in the transient and accident analysis of the Cycle 19 core as part of the NRC approved reload analysis methods. Cycle 19 operation will be operated within the design limits to protect fuel integrity. No plant modifications are required to accommodate the new core design or to operate Cycle 19. *The evaluation performed in support of EC64717 found that the RBS C19A core operating limits (including stability region boundaries) remain unchanged by the RBS C19C ULP design.* As previously described, the design and operating features of the GNF3 LUA are very similar to that of GNF2, such that there are no new failure modes created with its introduction into the RBS C19 core. No additional operating modes are required for the operation of Cycle 19. The loading of GNF3 LUA in the RBS C19 core design is such that they will operate with margin to thermal limits.

The introduction of NSF LUCs into the RBS core is not expected to result in any anomalous performance of the fuel channels; previous industry experience with NSF has shown corrosion resistance that is slightly improved over that of the Zircaloy-4 channels currently in the RBS core. The structural and physical properties of the NSF alloy is similar in terms of parameters such as stiffness and yield strength to that of the Zircaloy-4 alloy already in the RBS core. Thus its ability to perform the design function of a fuel channel (i.e., provides coolant flow path, guide for control rod travel, structural support for fuel in lateral loading, coolant flow leakage path at the core support plate which is important for in-core instrument function, transmits seismic loadings to the top guide and the fuel support/core internal structures, etc.) is similar to that of the fuel channels already present in the RBS core.

Therefore, the proposed activities will not create the possibility of an accident of a different type than any evaluated previously in the SAR.

6. **Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the USAR?** ☐ Yes ☒ No

BASIS:

The Cycle 19 reload core design cannot create the possibility of a malfunction of equipment important to safety with a different result than any previously evaluated. As previously described, the design and operating features of the GNF3 LUA are very similar to that of GNF2, such that there are no new failure modes created with its introduction into the RBS C19 core. Equipment important to safety will still be required to function in the same manner with the *Cycle 19C core* as with previous core designs. No plant modifications are required to accommodate the new fuel. The change in core characteristics does not change any parameter that would affect the function of equipment important to safety.

The *Cycle 19C ULP fuel* and GNF3 LUA are compatible with fuel that was inserted in previous cycles. The fuel will be operated within the design limits which will ensure all mechanical design criteria (e.g., centerline melt, clad strain) are met throughout the expected bundle lifetime. The introduction of the NSF LUCs is not expected to impact any of these design functions since the mechanical properties of the NSF alloy are very similar to those of Zircaloy-4. Previous experience at other BWRs (including BWR/6) of the NSF LUCs has demonstrated favorable performance over several cycles of continuous in-core exposure. Since the fuel designs meet all the fuel design acceptance criteria, the Cycle 19 core will not create a new mode of malfunction of equipment. Hence, introduction of the GNF3 LUA and NSF LUC into the Cycle 19 core will not increase the possibility of a malfunction of equipment important to safety with a different result than any evaluated in the SAR.

Therefore, Cycle 19C operation will not create the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the USAR.

7. Result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered? ☐ **Yes** ☒ **No**

BASIS:

Mechanical analyses have been performed to ensure that all fuel in the *Cycle 19C core* meets the mechanical design limits for steady-state operation as well as transient conditions including fatigue damage, creep collapse, corrosion, fuel rod internal pressure, rod bow, internal pressure, etc. This includes the GNF2 and GNF3 fuel. Additionally, no *Cycle 19C fuel* will exceed the applicable burn-up limits or thermal-mechanical integrity limits. The *C19C core design* ensures that the GNF3 LUA are operated with margin to the required operating limits.

Core operating limits have been developed using NRC approved methods to ensure that the *Cycle 19A fuel* will not exceed the MCPR safety limits for steady-state operation and Anticipated Operational Occurrences (AOO). Similarly, operating limits have been developed to ensure that the *Cycle 19A fuel* will not exceed the 1% cladding strain limit or experience core-wide fuel melt during steady-state operation or AOO's. *As previously described, vendor evaluations documented in COLR Ref.3.1.5 demonstrate, using NRC approved methods, that the C19A operating limits remain valid for operation of the C19C ULP core.*

As described in EC55796, a bounding pressurization event with a failure of the direct scram has been analyzed for Cycle 19 to ensure compliance with American Society of Mechanical Engineers (ASME) code requirements. This analysis indicates that the vessel pressure safety limit is not exceeded for Cycle 19. *As described in EC64717, the RBS C19C ULP design continues to satisfy the ASME code requirements, indicating that the vessel pressure safety limit is not exceeded for Cycle 19C.*

A design basis limit for the peak fuel enthalpy of 280 cal/gm has been established for the control rod drop accident (CRDA) to preclude significant fuel cladding failure such that core geometry and cooling may be impacted. An evaluation has demonstrated that the generic GNF CRDA analysis is applicable to *RBS Cycle 19C core operation*, with GNF3 LUA. This generic analysis shows that a CRDA will not exceed the 280 cal/gm peak enthalpy limit. Since this accident is a localized event and the peak enthalpy does not exceed 280 cal/gm, there is no impact on the vessel or containment pressures. As such their respective limits are not exceeded.

10CFR50.46 provides limits associated with the ECCS performance analysis (LOCA analysis). Two such limits are Peak Clad Temperature (PCT) and local clad oxidation. Although these limits are not subject to 10CFR50.59, they are discussed in this evaluation for completeness. River Bend specific analyses have been performed for GNF2 and GNF3 fuel in accordance with 10CFR50.46. These analyses, which are applicable to Cycle 19, show that the PCT and local oxidation are well below the limits set forth in 10CFR50.46. The core-wide metal water reaction, which is used to evaluate compliance with the containment design limit, is bounded by the C18 configuration. The containment

pressure design limit will not be exceeded in *Cycle 19C*. The GNF2 & GNF3 designs have no impact on LOCA performance of the fuel, as described by GNF in demonstrating compliance of the design with GESTAR-II licensing requirements.

An Anticipated Transient Without Scram (ATWS) evaluation was performed for GNF2 new fuel introduction at RBS in C17. As described in EC55796, this evaluation is applicable to Cycle 19 operation. *Since the RBS C19C ULP makes use of re-insert GNF2 fuel from the same RBS C17 reload batch fuel, the ATWS related impacts from use of GNF2 continue to apply to operation of Cycle 19C.* As previously described, the use of GNF3 fuel in the RBS C19 core was confirmed to not impact the existing ATWS core-wide analysis results. The fuel specific GNF3 ATWS results likewise remain well bounded by the ATWS event acceptance criteria. The evaluation demonstrates that the vessel pressure remains below the ASME emergency vessel pressure limit of 1500 psig and the containment parameters (suppression pool temperature, containment pressure) remain below applicable design limits for the ATWS event.

Additional cycle-independent evaluations have been reviewed or performed for Cycle 19 operation, including Appendix R (Fire Protection), hydrogen analyses, and Station Black Out (SBO). As described in EC55796, these evaluations are applicable to Cycle 19*A* and their respective acceptance limits are not exceeded. *As described in EC64717, these evaluations are applicable to Cycle 19C and their respective acceptance limits are not exceeded.*

Given the similarities of the physical performance of the NSF alloy with that of Zircaloy-4, introduction the NSF LUCs into the RBS core is not expected to alter in any way the predicted response of the fuel to any of the above described challenges. Previous industry experience with NSF LUCs has confirmed this expectation through several cycles of in-core exposure.

Based on the discussion above, no design basis limit for a fission product barrier as described in the USAR will be exceeded or altered.

8. Result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The *Cycle 19C* reload analyses performed by the fuel vendor utilized NRC approved methods as listed in Technical Specification 5.6.5, specifically GESTAR-II methodology, for GNF2 and GNF3 fuel described in the COLR and throughout the USAR. The GNF3 LUA and NSF LUC are introduced under the Lead Use application of the GESTAR-II methodology. Lead Use is a defined design method in GESTAR-II, thus the specific analyses required to support GNF3 LUA and NSF LUC have followed the method (GESTAR) required for RBS reloads. The GNF3 LUA and NSF LUC introduced into the RBS C19 design have met the required GESTAR-II acceptance criteria.

Therefore, Cycle 19C operation will not result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

I. OVERVIEW / SIGNATURES[1]

Facility: RBS **Evaluation # 2017-001 / Rev. #:** 0

Proposed Change / Document: EC69132 - Cycle 20 COLR (LAR-2017-02)

Description of Change: The core reload is a recurring activity for each fuel cycle. At the end of each fuel cycle, depleted fuel assemblies are discharged from the core and replaced by fresh reload assemblies. The remaining bundles resident in the core are shuffled to new locations and fresh fuel is loaded in accordance with the next cycle's core design and Reference Loading Pattern (RLP). This evaluation addresses the Cycle 20 reload changes for operation of the core including implementation of the GNF2.02 fuel design option. As described in the preceding PAD (EC69132), this 50.59 evaluation was completed to document the impact of proposed changes to the RBS Core Operating Limits Report (COLR) which reflect C20 core operation of the GNF2.02 fuel.

EC69132 addresses changes associated with the Cycle 20 reload core into which 212 fresh GNF2.02 bundles and 4 fresh GNF2 bundles were loaded during RFO19. All of the RL19 fuel bundles use NSF channels. The current COLR containing the Operating Limit Minimum Critical Power Ratio (OLMCPR) limits, Average Planar Linear Heat Generation Rate (APLHGR) limits, Linear Heat Generation Rate (LHGR) limits, APRM flow biased limits and stability region boundaries and the reference core loading pattern (RLP) is revised to reflect operation of the Cycle 20 core.

However, EC69132 and this 50.59 evaluation ONLY address RBS Cycle 20 operation from BOC to MOC conditions, as described in EC69132. Likewise, the RBS C20 COLR update is ONLY applicable from BOC to MOC; using BOC of 3/2/2017 and LF=1.00, a conservative estimate for MOC20 is 08/10/2018. This condition is being tracked for RBS C20 in CR-RBS-2016-06418. Operations from MOC20 thru EOC20 will be addressed separately in an evaluation of the engineering products of revised analyses being performed for RBS C20 application.

Summary of Evaluation: The BWR reload process requires that the vendor perform calculations to confirm the new core design meets the licensing acceptance criteria and the 'Analysis of Record' (AOR) continues to bound the reload core design. Therefore the core design features requiring the performance of reload transient analysis activity are considered adverse changes that must be evaluated under the 50.59 process.

The RBS C20 core design has been developed using NRC approved methods, General Electric Standard Application for Reactor Fuel (GESTAR-II, NEDC-24011-P-A-23), which is referenced in TS 5.6.5. The C20 core will be operated in conformance with cycle specific, generic, and fuel design specific analyses provided by Global Nuclear Fuel (GNF) using the GESTAR method. The C20 core operating limits will be incorporated into the core monitoring system prior to startup of C20. LAR-2017-01 was initiated to implement the required changes to the C20 COLR.

The fresh reload fuel for RBS C20 uses NSF channels, as approved for reload quantities in GESTAR-II. The C20 reload NSF channels are identical to the NSF LUC channels installed into RBS C19 (50.59 Evaluation # 2015-001 / Rev. #:0, OSRC Mtg# 15-004), which continue to operate in the C20 core. Likewise, the GNF3 Lead Use Assemblies (LUA) installed into RBS C19 also used NSF channels and will continue to operate in C20. NSF channels are expected to continue to improve RBS core performance with respect to mitigating the potential for channel distortion. Use of the RBS C20 reload NSF channels fully complies with all plant design and licensing basis criteria.

The RBS C20 reload fuel uses the GNF2.02 design option. EC69132 documents details of the design and its acceptability for RBS application. The vendor used their NRC approved fuel design method and documented that the GNF2.02 bundle meets all design and licensing criteria of GESTAR-II. Therefore, per Section 1.1 of the

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

GESTAR-II methodology for fuel and core design, since the GNF2.02 fuel design complies with all GESTAR-II design and licensing acceptance criteria, such compliance constitutes USNRC acceptance and approval of the fuel design without specific USNRC review.

There is no impact on fuel operating limits or licensing analyses as a result of the application of the GNF2.02 design. The resulting RBS C20 fuel specific operating limits are addressed in LAR-2017-01 as a part of the changes to the C20 COLR.

Transient analyses results for reload design include review of ASME overpressure response to confirm compliance with RBS TS 2.1.2 and TS Bases 2.1.2. The RBS C20 transient analyses have demonstrated that the TS requirements cannot be met for operations beyond MOC core exposure – using the current analysis method (ODYN). However, through application of another NRC approved vendor method (TRACG), the TS requirements can be met through EOC core exposure. Entergy has initiated a project for the fuel vendor to develop the TRACG model for RBS, and apply the results to RBS C20 core operating parameters. In the interim, using ODYN (the existing vendor method), the RBS C20 core has been shown to meet all design and licensing criteria (including ASME overpressure analyses) within operations from BOC to MOC.

Therefore, EC69132 and this 50.59 evaluation ONLY address RBS C20 operation from BOC to MOC; and the attached RBS C20 COLR is likewise restricted to use ONLY for operation from BOC to MOC. This condition is being tracked for RBS C20 in CR-RBS-2016-06418. The TRACG project is scheduled to complete well before MOC20 (8/10/2018) to support completion of an engineering evaluation and 50.59 to document the revised transient analyses results and incorporate them into a revised RBS C20 COLR – before core exposure reaches the limits of the ODYN analysis.

Background Information

Mechanical

RBS implemented an Enhanced Lead Use Channel program (per GNF Document MFN 12-074, Enhanced LUC Program for NSF Channels) in C19 (50.59 Evaluation #2015-001, OSRC Mtg# 15-004) to introduce the use of GNF's NSF alloy for fuel channels. RBG-47552 documents the notification to NRC of the LUC program implementation at RBS. 49 LUC channels and 4 Lead Use Assembly (LUA) channels, a total of 53 bundles in the C19 core, used the NSF channel material. All of these C19 bundles with NSF channels remain in the C20 core. The fresh reload fuel for RBS C20 also uses NSF channels, as approved for reload quantities in GESTAR-II; bringing the population of NSF channels in the RBS C20 core to a total of 269. The NSF LUC has extensive industry experience with exposures of channels up to 3 & 4 two-year cycles at several BWR units (including BWR/6). Monitoring and inspection results for these applications have demonstrated favorable performance. GGNS and RBS inspections of once-burned NSF LUCs have yielded acceptable performance. NSF channels are expected to continue to improve RBS core performance with respect to mitigating the potential for channel distortion. As previously documented for RBS C19 application, the NSF channel material meets all fuel design and licensing requirements defined in GESTAR-II; and thus requires no change to any description of the performance of the fuel in the RBS USAR. Although an increased channel bow condition in the remaining Zircaloy-4 channels can result in increased friction between the control blade and its corresponding fuel assemblies, control rod settle and insertion testing (OSP-0061) will continue to be performed during Cycle 20 per the guidance provided in fleet procedure EN-RE-216 to ensure that any increased axial friction loads on the channel and fuel assembly load chain remain below acceptable limits. Numerous previously irradiated C20 fuel assemblies (total of 62) were re-channeled with new Zircaloy-4 channels (identical replacements) to enhance the C20 core in terms of its resistance to channel distortion (i.e., reduce potential for channel bow).

The GNF2 fuel mechanical design has been reviewed for use at River Bend (50.59 Evaluation 2011-004, OSRC Mtg# 11-006). No unusual failure modes or increased failure frequency have been identified for this fuel design. This is the 4th cycle of operation for GNF2 fuel and this fuel design has accumulated operational experience at RBS and other plants (including Entergy BWR units) with no significant problems. The bundles will operate within the power history assumptions in the fuel mechanical analyses and will experience exposures within the analyzed burnup limits of the mechanical designs, including those bundles that will be irradiated for a third cycle. Mechanical design analyses [USAR 4.2.1] have been performed with NRC-approved methodology to evaluate

mechanical criteria including cladding steady-state strain and stresses, transient strain and stresses, fatigue damage, creep collapse, corrosion, hydrogen absorption, fuel rod internal pressure, etc. All parameters were found to meet their respective design limits for the Cycle 20 core.

The RBS C20 reload fuel uses the GNF2.02 design option for the reload fuel, and has been demonstrated to meet all applicable fuel and licensing criteria per the vendor's design methodology, GESTAR-II, which is listed in the approved methodologies for RBS in TS 5.6.5. The GNF2.02 design includes small modifications to the past GNF2 designs used in RBS C17-C19. The modifications were developed to enhance the debris capture potential for the debris filter used in the lower tie plate (LTP) and to reduce the debris capture potential for the spacer design. In addition, small changes were made to the configuration of the spacer to utilize a design with more robust seismic performance and greater support for the corner rod location. EC69132 documents details of the design and its acceptability for RBS application. The vendor has demonstrated, through predictive analysis and model testing, that the in-core performance of the GNF2.02 design matches that of previous the GNF2 design – such that the GNF2.02 features are considered "interchangeable" with the GNF2 counterparts. There is no impact on fuel operating limits or licensing analyses as a result of the application of the GNF2.02 design.

Nuclear

The neutronic characteristics of the Cycle 20 core design have been considered in the safety analysis. The nuclear characteristics of the reload GNF2 fuel remain unchanged from previous RBS reloads. Since the GNF2.02 fuel design option is interchangeable with previous GNF2 designs, there is no impact on the neutronic performance of the reload fuel (use of identical materials). As previously described for the C19 NSF LUC (50.59 Evaluation #2015-001, OSRC Mtg# 15-004), the nuclear characteristics of the NSF channels are very similar to that of currently used Zircaloy-4 fuel channels, no nuclear related impacts are expected from introduction of the NSF channels. The material properties of the NSF alloy are sufficiently similar to current channel materials such that NSF channels are expected to be dimensionally, structurally and mechanically compatible with Zircaloy-4 channels within a four bundle cell. Differences in the mechanical, thermal, and nuclear properties of NSF, as compared to Zircaloy-4, are minor and do not affect the methodology used to design and analyze the channels. Similarly, the minor alloy differences between NSF and Zircaloy have little to no significance to core design, transient, stability or ECCS-LOCA analyses. Adequate shutdown margin has been predicted by analysis and will be confirmed during startup tests per Technical Specification 3.1.1. In addition, the hold-down capability of the standby liquid control system has been confirmed. Therefore, the probability of inadvertent criticality has not been increased by the introduction of GNF2.02 and/or NSF reload channels in the Cycle 20 core.

Is the validity of this Evaluation dependent on any other change? ☐ **Yes** ☒ **No**

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ **Yes** ☒ **No**

Preparer: Jim Head / (See EC69132 for signature)/ ESI/ Fuel and analysis / 2/13/2017
Name (print) / Signature / Company / Department / Date

Reviewer: Melissa Litherland /(See EC69132 for signature)/ EOI/Nuclear Analysis / 2/26/2017
Name (print) / Signature / Company / Department / Date

OSRC: William L Carter / [signature] / 03/02/17
Chairman's Name (print) / Signature / Date

OSRC Meeting # 17-003

II. 50.59 EVALUATION

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. **Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR?** ☐ Yes ☒ No

BASIS:

RBS Cycle 20 operation will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR. The precursors to these events are independent of the core design and the frequency classifications reported in USAR Chapter 15 are unaffected by the core parameters. The following considerations support this conclusion.

Core Operating Limits

Operating Thermal Limits

Having followed approved design methods for the RBS C20 core design, power operation within the core operating limits (Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR), Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)) will ensure that the appropriate safety criteria are met during normal operations, Anticipated Operational Occurrences (AOO), or accidents. The precursors to these events are independent of the core design and the frequency classifications reported in SAR Chapter 15 are unaffected by the core parameters. Therefore, the core parameters in the COLR have no effect on the probability of the occurrence of any accident described in the SAR.

APRM Limits and Stability Region Boundaries

As with the normal reload review process for any core design, the stability performance of the reload design must be evaluated against various acceptance criteria specified in the E1A stability solution. Analyses have been performed to demonstrate that the revised E1A region boundaries for Cycle 20 met the channel and core decay ratios acceptance criteria. The change in E1A stability region boundaries implemented in the RBS C17 reload design will continue to avoid a thermal-hydraulic instability during the Cycle 20 operation. The previously described core design with NSF channels and GNF2.02 fuel has been analyzed and validated to meet all required stability performance criteria. Therefore, the probability of a thermal hydraulic instability has not increased.

Design Basis Accidents (Limiting Faults)

The design basis accidents analyzed in support of Cycle 20 are the control rod drop accident (CRDA), the fuel handling accident (FHA), and the loss of coolant accident (LOCA). The FHA is discussed separately.

The probability of the occurrence of design basis accidents is not dependent on the core configuration. No changes to the plant design are required for the Cycle 20 core. The Cycle 20 core loading will not affect the precursors to any of the Chapter 15 events, including LOCA analyses.

The Control Rod Drop Accident (CRDA) results from a failure of the control rod-to-drive mechanism coupling after the control rod becomes stuck in its fully inserted position. Although a channel bow condition can result in friction between the control blade and its corresponding fuel assemblies, analyses

have shown that there would not be sufficient friction to result in a mechanical failure of the coupling. Additionally, the control rod drive mechanism would not produce enough force to result in a mechanical failure of the coupling even if the channel bow was so severe that the assemblies would preclude blade movement. Channel bow associated with GNF2 reload fuel at high exposure is no more than the channel bow associated with previous RBS reload fuels (i.e., ATRIUM-10, GE14). The addition of NSF channels on reload fuel and the GNF2.02 design option is not expected to have any negative impact on channel performance in the core. The use of NSF for reload channels is now incorporated into the NRC approved design method for the vendor (GESTAR-II); based on industry experience with LUC programs with multiple cycles of exposure, the performance of the NSF channels has been similar to or improved over that of the Zircaloy-4 channels already present in the RBS core. As such, channel bow is not considered a precursor to the CRDA, and any increased bow associated with the high exposure fuel bundles would not increase the probability of this event.

Fuel Handling Accident and Fuel Loading Error

The FHA related impacts of GNF2 were previously evaluated for RBS application in RBS C17. As described in RBS C20 Fuel Receipt EC67910, the GNF2.02 design is identical in its overall physical characteristics to that of GNF2 and that its interface with handling equipment is likewise similar to GNF2, there is no expectation for an increase in the probability of occurrence of a FHA.

The fuel loading error (FLE) event involves improper loading of a fuel bundle into the core, related to human error associated with the ability to properly identify a bundle by its serial number identification and ability to determine the proper orientation of the bundle when loading into the core. These factors are driven by design features of the upper tie plate of the bundle, where the bundle ID is machined and orientation queues are built into the UTP design. The identification and orientation features of the GNF2.02 design have not changed from GNF2 fuel used previously at RBS. Thus, there is no increase in the probability of occurrence of an FLE event.

On these bases, the probability of occurrence of accidents previously identified in the SAR is not increased for the Cycle 20 core.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR? ☐ Yes ☒ No

BASIS:

No plant modifications are required to accommodate the Cycle 20 core design. Related specifically to the fuel in the reloaded core, the only additional loads placed on plant equipment would be due to changes in the characteristics of the fuel as it interfaces with other core components and functions; such as stability, decay heat, reactor internal pressure differences, reactor internal structural impacts, transient responses, neutron fluence, radiation source term, recirculation system impacts, changes to EOPs, fuel storage criticality, or other severe accident related affects. All of these topics were specifically addressed by the fuel vendor's assessment for introduction of the GNF2 design into the RBS core; and the use of GNF2.02 has been demonstrated to have no impact on any performance characteristics of the fuel. The GNF2.02 design features are considered 'interchangeable' with the GNF2 counterparts. The C20 reload evaluation has shown that the GNF2.02 bundles are compatible with co-resident fuel and core components in terms of its form, fit, and function. GNF2.02 does not introduce an increased potential for interference with core components (such as control rod blades, instrument tubes, etc.). The nuclear performance characteristics of the GNF2 design are not changed by the GNF2.02 design. Therefore, GNF2.02 introduction into the RBS C20 core does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of core components.

The only additional loads placed on plant equipment associated with the reload GNF2 fuel would be due to increased friction between the control blades and excessively bowed channels. As described in EC69132,

the reference loading pattern (RLP) design includes fuel channel designs previously used at RBS, and the re-channel of numerous previously irradiated fuel bundles in the RBS C20 core will reduce the potential for occurrence of excessive channel bow. Introduction of NSF channels on reload fuel does not impact previously evaluated SSC's or their malfunction as described in the USAR since its properties are very similar to those of the co-resident Zircaloy-4 fuel channels in the RBS core (as previously discussed in 50.59 Evaluation #2015-001, OSRC Mtg# 15-004). Based on previous experience with bowed fuel at RBS and other BWR-6 units, increased control blade friction can result in increased control rod settle times but is not expected to significantly impact scram times. Technical Specification scram time testing and control rod settle and insertion testing will continue to be performed during Cycle 20, in accordance with guidance provided in fleet procedure EN-RE-216. These actions would identify any potential scram time or other impacts such that appropriate corrective actions are taken. These actions will ensure that any increased control blade friction loads are not sufficient to cause any failures associated with the control blades or the control blade drive system, the fuel assembly load chain, or the vessel internals.

Analysis of Reactor Internal Pressure Differences (RIPDs) resulting from the introduction of a new fuel type with different hydraulic characteristics is done to ensure design basis criteria are met. The pressure across the RBS C20 core is the result of the dominant fuel in the core, GNF2. The previous C19 introduction of four GNF3 LUA in the 624 bundle RBS core has very little impact on the overall core hydraulic characteristics in the C20 core. In addition, the GNF3 hydraulic performance is bounded by past fuel types used at RBS. As previously described, the GNF2.02 design option was demonstrated to have no impact on the flow characteristics or hydraulic resistance of the GNF2 bundle; both the improved spacer and improved debris filter designs are considered 'interchangeable' with previous GNF2 counterparts.

A reactor internal structural assessment has been performed for the reload fuel. This assessment included normal, upset, emergency and faulted condition loads on the reactor internals and concluded that the reactor internals remain qualified for the reload fuel. The evaluation of The previous C19 introduction of four GNF3 LUA found no adverse impacts to the structural integrity of the reactor internals with regard to RIPD or seismic loading, and this result continues to apply for C20 operations. Based on evaluating the combined effects of RIPD, seismic, and fuel assembly weight, as stated previously for the C19, the introduction of four GNF3 LUA will have an insignificant effect on the structural integrity of the reactor internal components. Likewise, as previously described, the GNF2.02 bundles are identical to the previous GNF2 bundles (notwithstanding cycle specific changes in uranium loading). GNF2.02 introduction into C20 results in no change to the structural integrity of reactor internal components which was previously discussed for introduction of GNF2 in RBS C17 (5059 Evaluation # 2011-004/ Rev. #: 0, OSRC Mtg#11-006).

The dominant fuel type, GNF2, dictates the seismic behavior if the RBS fuel core response. As documented in RBS C19 (50.59 Evaluation #2015-001, OSRC Mtg# 15-004) introduction of the four GNF3 LUA does not have any significant impact on the mass and stiffness of the fuel in the RBS core, and does not change the fuel fundamental frequency. This remains true for the C20 core configuration. Likewise, the following properties were documented for GNF3 LUA: the design basis fuel lift loads remain valid for the four GNF3 LUA, the current design seismic qualification of the RBS core remains unchanged in supporting operation with four GNF3 LUA. These properties also remain true for the C20 core configuration. The GNF2.02 design does not impact the mass and stiffness of the GNF2 fuel previously evaluated for seismic performance as part of its initial introduction into RBS C17 (5059 Evaluation # 2011-004/ Rev. #: 0, OSRC Mtg#11-006).

A conservative vessel over-pressurization analysis has been performed, which shows that the vessel pressure limit is not exceeded in operation from BOC20 to MOC20. The results of the analysis were reported in the Supplemental Reload Licensing Report (SRLR) for the RBS C20 core, in which the GNF2.02 reload fuel and GNF3 LUA were explicitly modeled. The RBS C20 core with GNF2.02 and GNF3 LUA was shown to continue to meet the required ASME overpressure protection criteria up to MOC20. Therefore, this 50.59 evaluation only applies to RBS C20 operation from BOC to MOC (see CR-RBS-2016-06418-CA3).

The impact on the vessel fluence associated with the RBS C20 reload was evaluated in EC69132. It was found that the current RBS vessel fluence analysis remains valid with the C20 reload core design. Given the interior location of the GNF3 LUA in the RBS C20 core with respect to the vessel wall, GNF3 and GNF2.02 bundles do not impact vessel fluence in operation of the C20 core and the current RBS P/T curve remains applicable. Thus, use of GNF3 and GNF2.02 fuel in the RBS C20 core do not result in an increase in the likelihood of occurrence of fluence induced damage to the vessel.

The containment response is determined primarily by fuel-independent parameters such as RHR heat exchanger performance, ECCS coolant injection flow, service water temperature, and initial suppression pool temperature. As previously described, introduction of the GNF2.02 reload fuel and continuation of GNF3 LUA does not result in more than a minimal impact on the containment response.

With regards to post-LOCA hydrogen control, per 10CFR50.44 RBS must have a system that will mitigate hydrogen from a metal-water reaction involving 75% of the core cladding surrounding the active fuel region. The containment igniter system meets this requirement for the current GNF2 reload core. Introduction of the GNF3 LUA does not have more than a minimal impact post-LOCA hydrogen control.

The precursors to any malfunction of equipment important to safety are not affected by the Cycle 20 reload core. Therefore, there is not more than a minimal increase in the likelihood of an occurrence of a malfunction of a SSC important to safety previously evaluated in the USAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR? ☐ **Yes** ☒ **No**

BASIS:

The acceptance criteria reported in USAR Section 15.0.3.1 and the Technical Specifications are satisfied for each event classification. Core operating limits have been developed to ensure that moderate frequency events do not violate the MCPR safety limit or fuel cladding strain limits. The consequences of infrequent events have been shown to meet the appropriate acceptance criteria while the individual acceptance criteria for the limiting faults have been demonstrated to be satisfied. Satisfying these acceptance criteria ensures an upper bound on the potential release of radiological source term, thus limiting the resulting consequences of the event. As such, the consequences of infrequent events and limiting faults described in the USAR are unchanged for the Cycle 20 reload core. The following considerations support these conclusions.

Anticipated Operational Occurrences (AOO) Events

The Cycle 20 core operating limits have been developed with NRC-approved methodologies such that the MCPR safety limit and the fuel cladding strain limit will not be violated by any analyzed moderate frequency transient initiated from any statepoint available to RBS. As such, no fuel failures are expected to result from any moderate frequency event. Thus the radiological source term and the potential extent of its release from the fuel have not been increased with the introduction of GNF2.02 in the RBS C20 core. These analyses considered RBS-specific operational modes such as Maximum Extended Operating Domain (MEOD), Single Loop Operation (SLO), Feedwater Heater(s) out of Service (FHOOS), Final Feedwater Temperature Reduction (FFTR), Pressure Regulator Out Of Service (PROOS), Main Turbine Bypass Out Of Service (MTBOOS) and End Of Cycle –Recirculation Pump Trip (EOC-RPT) inoperable. These core operating limits consist of MCPR and LHGR curves that are functions of flow, power, and exposure for both GNF2 and GNF3 fuel. These core operating limits have been incorporated into the core monitoring system. These limits consider conservative channel bow assumptions that bound the current measured bow data and the expected increased

bow associated with the highly exposed fuel. Introduction of the NSF is expected to improve the performance of the fuel channels. Industry experience with other applications of NSF LUC has yielded favorable results. GNF3 LUA loaded into the RBS C20 core will operate with margin to the thermal limit requirements. Approved methods were used to model the GNF3 and GNF2.02 bundles and confirm they respond in a similar manner to the co-resident GNF2 assemblies. Thus use of the GNF3 and GNF2.02 fuel in the RBS C20 core design does not result in an increase in the consequences any AOO event.

Infrequent Events

The fuel loading error (*i.e.*, misoriented and mislocated) is considered an infrequent event. The consequence of this event has been evaluated in accordance with the GESTAR requirements and shown to meet the respective acceptance criteria. Radiological analyses using the alternative source term (AST) have been performed to ensure that these events will not result in an increase in offsite or control room dose or doses greater than their respective acceptance criteria. Evaluation of the RBS C20 reload core design, including GNF3 and GNF2.02 demonstrated that the current radiological source term for RBS remains bounding. The radiological analyses reflect operation as described in the USAR.

Limiting Faults

The limiting faults at RBS include the fuel handling accident, the control rod drop accident, and the design basis LOCA. The radiological analyses for these events have been developed as part of the AST effort assuming certain core parameter values such as radial peaking factor and fuel burnup. Core parameters assumed in the dose analysis were reviewed and found to be bounding for the Cycle 20 core. The number of rod failures in a CRDA or FHA using GNF2 fuel remains bounded by the assumption used in the dose analyses. The Cycle 20 core (with inclusion of the GNF3 and GNF2.02) parameters have been shown to be bounded by the RBS AST source term and radiological analyses. For the Loss Of Coolant Accident (LOCA), MAPLHGR operating limits and single-loop multipliers have been developed for the Cycle 20 core configuration, including GNF3 & GNF2.02, such that the requirements of 10CFR50.46 are satisfied. The containment response for the Cycle 20 core, including GNF3 and GNF2.02 fuel, was found to not result in more than a minimal impact on previous GNF2 design based containment system response analysis. Review of the potential degraded core hydrogen generation has shown that the RBS C18 core configuration remains bounding with respect to the Cycle 20 core configuration, including GNF3 & GNF2.02 bundles. Review of the Cycle 20 core has shown that there will not be more than a minimal impact on the seismic/LOCA response. The GNF2.02 design has no effect on the LOCA and Transient Analyses and the consequence from these events is not impacted.

Compliance with licensing acceptance criteria for the CRDA is assured through adherence to the Banked Position Withdrawal Sequence (BPWS) where analyses have generically demonstrated large margin to licensing limits on enthalpy insertions. Given the similarities of the GNF3 and GNF2 bundles' nuclear characteristics, and that the GNF2.02 design is interchangeable with GNF2, the generic analysis remains applicable to RBS operation with both fuel types.

With respect to LOCA performance of the RBS C20 core the GNF3 bundles will be restricted to a peak linear heat generation rate (PLHGR) equivalent to that of GNF2. Since the GNF2.02 bundles are equivalent to previous GNF2 fuel in their form, fit, and function, the acceptance criteria of 10CFR50.46 remain satisfied in the C20 core configuration.

The FHA was analyzed previously for introduction of GNF2 in RBS C17. The analysis was performed for GNF3 bundles in the RBS core design. Given that the GNF3 LUA are loaded in non-limiting locations of the C20 core, they will accumulate less bundle exposure and less bundle source activity. Despite this loading restriction, the analysis applied the same conservative assumptions regarding exposure and radial peaking factor, which increased the assumed bundle source activity more than that associated with the slight increase in fuel mass of GNF3 versus GNF2 bundles. The results of the FHA analysis demonstrated that even given these conservative assumptions, the consequence multiplier of the GNF3 fuel remains

less than unity and is therefore bounded by the consequence of a FHA involving GNF2 fuel, confirming that the GNF3 LUA have no impact on the dose consequences of the FHA. The vendor demonstrated that the GNF2.02 design option is interchangeable with the previous GNF2 design; thus, introduction of GNF2.02 has no impact on the previous conclusions for FHA related to operation of GNF2 fuel.

The core radiation source term impact from the introduction of GNF3 is inconsequential since its design is similar to that of GNF2. As previously described, the C20 reload parameters affecting the radiological source term were compared against the acceptance criteria and found to remain bounded. Therefore, GNF3 LUA in the C20 core are expected to have no significant impact on the RBS core radiological source term. Radionuclide concentrations in reactor water and steam are not dependent on fuel type but on operating conditions of the core, and since GNF3 will not affect core performance; it will not affect coolant radiation sources. Since GNF2.02 is interchangeable with the GNF2 fuel design, there is no impact on the expected radiological source term with use of GNF2.02 in the RBS C20 core. Therefore, GNF3 and GNF2.02 fuel designs will not affect the dose consequences of postulated reactor coolant system accidents such as Main Steam Line Break.

A recirculation pump seizure event while in single loop operation is also a limiting fault, but is conservatively evaluated against the AOO acceptance criteria. As the AOO acceptance criteria (MCPR) are satisfied, the limiting fault acceptance criteria are satisfied. This was demonstrated for the introduction of GNF2 in RBS C17. The use of four GNF3 LUA in RBS C20 will not affect the results of the analysis due to the large conservatisms assumed in the calculation of the void coefficient. For the GNF3 LUA, the same critical power correlation as used for GNF2 (GEXL17) has been conservatively adjusted for application to GNF3. Likewise, given that the GNF2.02 design option is interchangeable with the previous GNF2 design, the performance of GNF2 fuel in this event is not impacted by its use in RBS C20. Therefore, the previously determined GNF2 SLO pump seizure limits apply to GNF3 and GNF2.02.

Based on the review above, it is concluded that the Cycle 20 reload design will not increase the consequences of an accident previously evaluated in the USAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR? ☐ **Yes** ☒ **No**

BASIS:

All equipment important to safety will function in the same manner with the Cycle 20 reload core as with the previous core design. The proposed reload does not require new hardware or modification of existing hardware. Any malfunction of key plant components as described in USAR which could impact fuel integrity has been factored into the Cycle 20 transient and accident analyses. The consequences of these malfunctions have been shown to meet their respective acceptance criteria and to remain unchanged and bounded for Cycle 20 operation. Specific to the Cycle 20 reload design with the use of GNF3 LUA and GNF2.02, it has been shown to be compatible with the other co-resident GNF2 fuel in the core. GNF2 fuel was introduced at RBS in C17, and it has demonstrated acceptable performance in reload evaluations and operations for C17 thru C19. As previously described, the performance of GNF3 as determined using approved methods is predicted to be similar to that of GNF2. Introduction of the NSF reload channels is compatible with the Zircaloy-4 channels currently used in the RBS core, and is expected to provide comparable or improved performance during operation. Specific topics are discussed below:

Core Operating Limits

The SLMCPR and AOO analyses are based upon Cycle 20 core design to determine operating thermal limits, using the NRC approved methodology. Operation within these limits during normal steady state operation provides assurance that MCPR safety limits and fuel thermal/mechanical design limit are not exceeded due to the most severe AOO event. An AOO event assumes a single equipment failure such as load reject or turbine trip which causes a spike in reactor power and heat flux. The transient analyses have included such severe events so that safety limits are always maintained. The applicability of the current

stability region boundaries depicted in the COLR has been confirmed for Cycle 20. As such, no fuel failures are expected to result from any AOO event. The loading the GNF3 LUA has been designed such that they will not be the limiting fuel in the core at any time in C20 operations and will operate with margin to the operating limits. The use of GNF2.02 has been demonstrated to have no impact on core performance as compared with GNF2. Thus the RBS C20 core is designed such that the potential release of bundle source radionuclides remains bounded by the limiting source term analysis. Therefore, the consequences of a malfunction of a safety-related structure, system or component have not increased.

Design Basis Accident

The design basis accidents reviewed in support of Cycle 20 are the control rod drop accident (CRDA), the fuel handling accident (FHA), and the loss of coolant accident (LOCA). The core parameters assumed in the RBS dose analyses remain bounding with respect to the Cycle 20 operating core parameters with use of GNF3 LUA and GNF2.02 reload fuel.

Therefore, Cycle 20 operation will not increase the consequence of a malfunction of equipment important to safety evaluated in the SAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the USAR? ☐ **Yes** ☒ **No**

BASIS:

The Cycle 20 GNF3 fuel has been shown compatible with GNF2 fuel that was inserted in previous cycles. In addition, the GNF2.02 fuel design has been shown to be interchangeable with previously used GNF2 fuel. There are no changes in the operation of any system; highly exposed fuel bundles are replaced with fresh bundles. The malfunction of key components as described in the USAR are analyzed in the transient and accident analysis of the Cycle 20 core as part of the NRC approved reload analysis methods. The RBS Cycle 20 core will be operated within the design limits to protect fuel integrity. No plant modifications are required to accommodate the new core design or to operate Cycle 20. As previously described, the design and operating features of the GNF3 LUA are very similar to that of GNF2, and those of GNF2.02 are identical to GNF2; such that there are no new failure modes created with their use in the RBS C20 core. No additional operating modes are required for the operation of Cycle 20. The use of GNF3 LUA and GNF2.02 reload fuel in the RBS C20 core design is such that they will operate with margin to thermal limits.

The use of NSF reload channels in the RBS core is not expected to result in any anomalous performance of the fuel channels; previous industry experience with NSF has shown corrosion resistance that is slightly improved over that of the Zircaloy-4 channels currently in the RBS core. The structural and physical properties of the NSF alloy was previously shown (50.59 Evaluation #2015-001, OSRC Mtg# 15-004) to be similar in terms of parameters such as stiffness and yield strength to that of the Zircaloy-4 alloy already in the RBS core. Thus its ability to perform the design function of a fuel channel (i.e., provides coolant flow path, guide for control rod travel, structural support for fuel in lateral loading, coolant flow leakage path at the core support plate which is important for in-core instrument function, transmits seismic loadings to the top guide and the fuel support/core internal structures, etc.) is similar to that of the fuel channels already present in the RBS core.

Therefore, the proposed activities will not create the possibility of an accident of a different type than any evaluated previously in the SAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the USAR? ☐ **Yes** ☒ **No**

BASIS:

The Cycle 20 reload core design cannot create the possibility of a malfunction of equipment important to safety with a different result than any previously evaluated. As previously described, the design and operating features of the GNF3 LUA are very similar to that of GNF2, and those of GNF2.02 are identical, such that there are no new failure modes created with their use in the RBS C20 core. Equipment important to safety will still be required to function in the same manner with the Cycle 20 core as with previous core designs. No plant modifications are required to accommodate the new fuel. The change in core characteristics does not change any parameter that would affect the function of equipment important to safety.

The Cycle 20 reload fuel and GNF3 LUA are compatible with fuel that was inserted in previous cycles. The fuel will be operated within the design limits which will ensure all mechanical design criteria (e.g., centerline melt, clad strain) are met throughout the expected bundle lifetime. The use of the NSF reload channels is not expected to impact any of these design functions since the mechanical properties of the NSF alloy are very similar to those of Zircaloy-4. Previous experience at other BWRs (including BWR/6) of the NSF LUCs has demonstrated favorable performance over several cycles of continuous in-core exposure. Since the fuel designs meet all the fuel design acceptance criteria, the Cycle 20 core will not create a new mode of malfunction of equipment. Hence, use of the GNF3 LUA, GNF2.02 reload fuel, and NSF reload channels in the Cycle 20 core will not increase the possibility of a malfunction of equipment important to safety with a different result than any evaluated in the SAR.

Therefore, Cycle 20 operation will not create the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the USAR.

7. Result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered? ☐ **Yes** ☒ **No**

BASIS:

Mechanical analyses have been performed to ensure that all fuel in the Cycle 20 core meets the mechanical design limits for steady-state operation as well as transient conditions including fatigue damage, creep collapse, corrosion, fuel rod internal pressure, rod bow, internal pressure, etc. This includes the GNF2 and GNF3 fuel. Additionally, no Cycle 20 fuel will exceed the applicable burn-up limits or thermal-mechanical integrity limits. The C20 core design ensures that the GNF3 LUA and GNF2.02 reload fuel are operated with margin to the required operating limits.

Core operating limits have been developed using NRC approved methods to ensure that the Cycle 20 fuel will not exceed the MCPR safety limits for steady-state operation and Anticipated Operational Occurrences (AOO). Similarly, operating limits have been developed to ensure that the Cycle 20 fuel will not exceed the 1% cladding strain limit or experience core-wide fuel melt during steady-state operation or AOO's.

As described in EC69132, a bounding pressurization event with a failure of the direct scram has been analyzed for Cycle 20 to ensure compliance with American Society of Mechanical Engineers (ASME) code requirements. This analysis indicates that the vessel pressure safety limit is not exceeded for Cycle 20 ONLY for operation from BOC to MOC.

As previously described (CR-RBS-2016-06418), the ASME overpressure analyses cannot meet the acceptance criteria in MOC to EOC operations without a change in the transient analysis model. Thus, a separate project has been launched to implement the use of a different NRC approved method for RBS C20 and provide revised COLR inputs prior to RBS MOC20. A separate engineering evaluation and 50.59 will be prepared to address implementation of the revised safety analyses and the associated COLR update for RBS MOC20 thru EOC20.

A design basis limit for the peak fuel enthalpy of 280 cal/gm has been established for the control rod drop accident (CRDA) to preclude significant fuel cladding failure such that core geometry and

cooling may be impacted. An evaluation has demonstrated that the generic GNF CRDA analysis is applicable to RBS Cycle 20 core operation, with GNF3 LUA and GNF2.02 reload fuel. This generic analysis shows that a CRDA will not exceed the 280 cal/gm peak enthalpy limit. Since this accident is a localized event and the peak enthalpy does not exceed 280 cal/gm, there is no impact on the vessel or containment pressures. As such their respective limits are not exceeded.

10CFR50.46 provides limits associated with the ECCS performance analysis (LOCA analysis). Two such limits are Peak Clad Temperature (PCT) and local clad oxidation. Although these limits are not subject to 10CFR50.59, they are discussed in this evaluation for completeness. River Bend specific analyses have been performed for GNF2 and GNF3 fuel in accordance with 10CFR50.46. These analyses, which are applicable to Cycle 20, show that the PCT and local oxidation are well below the limits set forth in 10CFR50.46. The core-wide metal water reaction, which is used to evaluate compliance with the containment design limit, is bounded by the C18 configuration. The containment pressure design limit will not be exceeded in Cycle 20. The GNF2.02 & GNF3 designs have no impact on LOCA performance of the fuel, as described by GNF in demonstrating compliance of the design with GESTAR-II licensing requirements.

An Anticipated Transient Without Scram (ATWS) evaluation was performed for GNF2 new fuel introduction at RBS in C17. As described in EC69132, this evaluation is applicable to Cycle 20 operation since the GNF2.02 design is shown to be interchangeable with previous GNF2 designs. As previously described, the use of GNF3 fuel in the RBS C20 core was confirmed to not impact the existing ATWS core-wide analysis results. The fuel specific GNF3 ATWS results likewise remain well bounded by the ATWS event acceptance criteria. The evaluation demonstrates that the vessel pressure remains below the ASME emergency vessel pressure limit of 1500 psig and the containment parameters (suppression pool temperature, containment pressure) remain below applicable design limits for the ATWS event.

Given the similarities of the physical performance of the NSF alloy with that of Zircaloy-4, use of the NSF reload channels in the RBS core is not expected to alter in any way the predicted response of the fuel to any of the above described challenges. Previous industry experience with NSF LUCs has confirmed this expectation through several cycles of in-core exposure.

Based on the discussion above, no design basis limit for a fission product barrier as described in the USAR will be exceeded or altered.

8. Result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The Cycle 20 reload analyses performed by the fuel vendor utilized NRC approved methods as listed in Technical Specification 5.6.5, specifically GESTAR-II methodology, for GNF2 and GNF3 fuel described in the COLR and throughout the USAR. The GNF3 LUA and prior NSF LUC were introduced under the Lead Use application of the GESTAR-II methodology in RBS C19. Lead Use is a defined design method in GESTAR-II, thus the specific analyses required to support GNF3 LUA and NSF LUC have followed the method (GESTAR) required for RBS reloads. The application of the required methods for the GNF3 LUA and NSF LUC used in the RBS C20 design has met GESTAR-II acceptance criteria.

Therefore, Cycle 20 operation will not result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

ATTACHMENT 1

SAFETY EVALUATIONS ASSOCIATED WITH

OTHER

SAFETY EVALUATION NUMBER: **None**

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ATTACHMENT 2

FSAR ONLY SUMMARY REPORT

ATTACHMENT 2

FSAR-ONLY EVALUATIONS

LBDCR NUMBERS: **08.02-022**

(TYPICAL)

I. LBDCR INITIATION

Thomas Moffitt	DE Electrical	4316	1	9/3/2015	08.02-022
INITIATOR'S NAME *(print or type)*	**DEPARTMENT**	**PHONE**	**UNIT**	**DATE**	**LBDCR #**

DESCRIPTION OF THE CHANGE (Attach additional pages if necessary; may also reference PAD Form.)
Delete reference to 48VDC system in Fancy Point Substation from Section 8.2 per CR-RBS-2015-05977.

	LICENSING DOCUMENT(S) AFFECTED	**AFFECTED SECTION/PAGE(S)** *(Attach marked-up pages)*
☐	**Operating License (OL)**	
☐	**Technical Specifications (TS)**	
☐	**Environmental Protection Plan (EPP)**	
☐	**Anti-Trust Conditions (Appendix of OL)**	
☐	**NRC Orders**	
☒	**Updated Final Safety Analysis Report (UFSAR)**	**Section 8.2 Pages 8.2-3,4, & 6**
☐	**TS Bases**	
☐	**Technical Requirements Manual (TRM) (including TRM Bases)**	
☐	**Quality Assurance Program Manual (QAPM)**	
☐	**Security Plan**	
☐	**Emergency Plan (EP)**	
☐	**Offsite Dose Calculation Manual (ODCM)**	
☐	**Spent Fuel Storage Cask Final Safety Analysis Report (CFSAR)**	
☐	**Spent Fuel Storage Cask Certificate of Compliance (CoC)**	
☐	**Spent Fuel Storage Cask CoC Bases**	
☐	**10 CFR 72.212 Evaluation Report (212 Report)**	
☐	**Fire Protection Program (FPP)/Fire Hazards Analysis (FHA)**	
☐	**Core Operating Limits Report (COLR)**	
☐	**Other (Specify)** _____	

METHOD(S) ALLOWING THE CHANGE			
☐	**PAD Review** (Attach a copy)	☐	**10 CFR 50.48 / EN-DC-128 Review** (Attach a copy)
☐	**10 CFR 50.59 Evaluation** (Attach a copy)	☐	**10 CFR 50.54 Review** (Attach a copy)
☐	**10 CFR 72.48 Evaluation** (Attach a copy)	☐	**Environmental Evaluation** (Attach a copy)
☐	**Approved NRC Change** (Attach a copy of NRC Letter or reference NRC letter number)	☐	**Editorial Change** (LBDs controlled under 50.59 or 72.48, only)
☐	**NRC Approval is Required**	☐	**Other Approval** (Attach a copy of supporting documents)
☒	**"UFSAR-only" Change (NEI 98-03)** **Check the appropriate box below:** ☐ **Reformatting** ☐ **Replacing Detailed Drawing** ☐ **Referencing other Documents** **Check the appropriate box below and provide a basis for removing information, if applicable:** ☐ **Removing Excessive Detail** ☒ **Removing Obsolete Information** ☐ **Removing Redundant Information** ☐ **Removing Commitments** **Removal Basis:** 48 V battery and DC system have been removed from Fancy Point. Transmission has replaced with 125VDC equipment. These items are not under RBS design control.		

II. LBDCR IMPLEMENTATION[1]

ACTIONS SUPPORTING IMPLEMENTATION			
LBD SECTION	REQUIRED ACTIONS		ACTION TAKEN OR TRACKING METHOD
	ACTION	RESP. DEPT	
8.2	Update text	Licensing	AR 00233838

III. LBDCR REVIEW AND APPROVAL[1]

REVIEW AND APPROVAL of LBDCR *(see Attachment 9.2)*		
Department	**Approved[2]**	**Date**
UFSAR Section Owner[3]	GREG SVESTKA	9/15/15
Peer Review[4]		9/16/15
LBD Owner	Kristi Huffstatler	9/17/15

1 Add additional table rows as needed.

2 The printed name should be included on the form when using electronic means for signature. Signatures may be obtained via electronic processes (e.g., PCRS, ER processes, Asset Suite signature), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail, attach it to this form.

3 UFSAR Section Owners should refer to EN-LI-113-01, "Updated Final Safety Analysis Report Change Process," for review expectations. N/A if change does not update the UFSAR.

4 Administrative peer review intended to verify changes have been incorporated correctly into revised LBD prior to issuance.

A fault of any section of the 230-kV bus is cleared by the adjacent breakers and does not interrupt operation of any of the remaining parts of the 230-kV switchyard bus. Only that element connected to the faulted section is interrupted.

The 500-kV bays of the substation are located northwest of and adjacent to the 230-kV bays and are constructed of SF_6 components. The 500-kV bays are arranged in a folded breaker-and-a-half scheme, consisting of six 500-kV GCB positions. The initial installation for Unit 1 consists of three circuit breakers and three 500-kV lines in a ring bus configuration. Fig. 8.1-7 illustrates the initial and final 500-kV configurations. There are two separate SF_6 charging systems for the 500-kV bays: one to serve the 500-kV SF_6 circuit breakers, and one to serve the 500-kV SF_6 buses, disconnecting switches, and air bushings. The initial ring bus configuration provides for the isolation of any faulted line without affecting the operation of any other line. It also provides for the isolation of any one breaker in the 500-kV SF_6 bus for inspection or maintenance without affecting the operation of any of the connecting lines or any other connections to the buses. The 500-kv buses terminate at the 500-kv SF_6 air bushings. Connections are made to the 500-kV grid and to the 230-500-kV transformers via air-insulated, outdoor-constructed bus work and overhead lines from these air bushings.

•→7

The ac auxiliary power requirements of the 230-kV and 500-kV bays are provided by two 750-kVA, 13.8-kV to 480-V oil-filled transformers supplied from onsite 13.8-kV buses 1NPS-SWG1A and 1NPS-SWG1B.

7←•

The dc requirements for the Fancy Point Substation relay and control systems are provided by two 125-V batteries ~~and one 48-V battery.~~ Each battery system is supported by its own charger which is provided power from the auxiliary ac power system and by an existing source external to River Bend Station.

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•→16
Control functions between the plant and the substation are provided by two diverse methods. Control cables are routed in a concrete-encased duct bank to the substation control house. Routing within the substation between the various relay panels and control equipment is accomplished via a protected cable trench. An optic cable underbuilt on the reserve station service steel pole lines provides another diverse method of transmitting control functions and information between the plant and substation. The optical information is decoded at the substation and forwarded to the appropriate piece of equipment via control cables routed in a cable trench or raceway that is physically separated by 5 ft or more from the other trench or raceway described herein. The routing separation is maintained over the route length except at termination points where the cables route to the same piece of equipment.
16←•

The 125-V battery system furnishes the control power for circuit breakers in both the 500-kV and 230-kV switchyard bays. A complete loss of both 125-V battery systems, including the battery charger, prevents the operation of all circuit breakers in the switchyard. The loss of the battery system in conjunction with a fault in the switchyard or any incoming line would require the operation of backup relaying elsewhere within the grid to clear the fault. Offsite power will be manually restored by isolating one of the reserve station service lines to an unfaulted line in the event of severe battery damage. The estimated time to perform the subject operation is 15 min after personnel arrive at the switchyard. LBDCR 08,02-22

~~The 48-V battery system provides operations voltage for the 500-kV static relaying.~~

•→16
The battery systems are monitored remotely using a SCADA (Supervisory Control and Data Acquisition) system which provides a low-voltage alarm to the Southwest-Transmission Operation Center (SWTOC) dispatcher in the event of malfunction. Additionally, the batteries receive a visual inspection weekly and a complete inspection for operability to manufacturer's specifications each 6 months. The weekly inspection consists of checking the electrolyte levels, the battery voltage, and the charge rate. The 6-month inspection includes checking the voltage and specific gravity of each cell, cleaning and retorquing the battery connectors, and if needed, the application of an equalize charge for about 24 hours. A form containing each of the above-mentioned items is filled out for each inspection. Completed inspection forms are kept on file at the Baton Rouge Substation Department.
16←•

All 230-kV circuit breakers are equipped with two independent trip coils and breaker failure protection for redundant power circuit protection. All of the protective relay systems for the 230-kV bays are redundant. These systems are overlapping so that each high-voltage component is covered by at least two sets of protective relays. The primary and the backup relay systems are supplied from separate current inputs, separate dc circuits from each 125-V battery, and are connected to separate trip coils of

LBDCR 08-02-022

relays. ~~The two primary static relay systems have separate current inputs and receive power from the 48-V battery system which acts as an isolated power supply.~~ The electromechanical backup relay system has separate current inputs and receives its power from one of the redundant 125-V battery systems. Cross tripping between the trip coils is used. The potentials for the primary and backup relay systems associated with the 500-kV lines are provided from one set of coupling capacitor voltage transformers on each line terminal. The secondary potentials are separated into two systems of junction boxes in the switchyard and are treated as redundant systems from this point. The potentials for the 500 to 230-kV transformer backup relaying (500 kV) are provided from one set of bushing potential devices on the 500-kV side of the transformer.

The primary relay systems for the two 500-kV lines are: 1) phase comparison relaying over a CS26 power line carrier channel; and 2) directional comparison tripping with phase and ground distance relays using a frequency shift audio tone, modulated on a microwave channel. The backup relay system for the two 500-kV lines is a three-zone distance phase and ground relay system that initiates local tripping.

•→16 16←•

The primary relay system for the 500-kV line to the 500 to 230-kV transformer is a separate restraint bus differential system. The primary relay system for the 500 to 230-kV transformer is a separate restraint transformer differential system. The backup relay system (500 kV) for the 500 to 230-kV transformer is a single zone distance phase with directional overcurrent ground relaying that initiates local tripping.

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ATTACHMENT 3

COMMITMENT CHANGE SUMMARY REPORTS

ATTACHMENT 3

COMMITMENT CHANGE SUMMARY REPORTS

COMMITMENT NUMBERS:	**P-14125**
	P-15524
	P-15532
	P-15533
	P-15746
	P-15747
	P-15748
	P-17156

LRS/CMS I.D. See Att. 1 Site Reg. Assurance Tracking Number: N/A

Source Document / Date: See Att. 1

Commitment:
- ☐ Cancellation
- ☒ Historic/Hist/Retired/NEI
- ☐ Revision
- ☐ Superseded

Has the original commitment been implemented? ☒ YES ☐ NO, stop and notify site Regulatory Assurance

Original Commitment Description:

Multiple (see LRS)

Revised Commitment Description:

N/A

Summary of Justification for Change (attach additional sheets if necessary):

The attached commitments listed in fleet procedure EN-DC-115, "Engineering Change Process" were reviewed as part of a Nuclear Promise initiative and based on the review, it was determined that these commitments have been tracked for more than two years and are well established in work processes. Therefore, it was determined that these commitments may be removed from continuing tracking.

Prepared By: Eve Clevenger Date: 4-17-17
Print Name / Signature

Management Approval: James G. Rogers Date: 4-17-17
Print Name / Signature

Forward completed form to site Regulatory Assurance.

Reg. Assurance Management (or designee) Concurrence: Kristi Huffstatler Date: 4-17-17
Print Name / Signature

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program Manual, Fire Protection Program, or Security Plan?

☐ YES STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54, 10 CFR 50.59, 10 CFR 50.55) to evaluate commitment

☒ NO Go to Part II.

PART II

2.1 Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its specified safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its specified safety function?

☐ YES Go to Question 2.2.

☒ NO Continue with Part III. Briefly describe rationale: *These commitments relate to historical administrative Engineering procedure updates/enhancements and have no impact on safety functions of SSCs, or the ability of plant personnel to ensure the SSC is capable of performing its specified safety function.*

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

- Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ YES ☐ NO Describe basis below

- Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ YES ☐ NO Describe basis below

- Does the revised commitment involve a significant reduction in a margin of safety?

☐ YES ☐ NO Describe basis below

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain necessary approvals. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required. IF all three questions are answered NO, THEN go to Part III. (Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an obligation (e.g., rule, regulation, order or license condition)?

☒ YES Go to Question 3.2.

☐ NO Continue with Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☒ YES Briefly describe rationale (attach additional sheets as necessary). IF commitment not yet implemented, THEN notify NRC of revised commitment date prior to the original date in which the commitment was to be completed/satisfied. *These are historical commitments which were previously implemented, and thus do not involve any revisions to commitment dates.*

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☒ YES Go to Question 4.2.

☐ NO Continue with Part V.

4.2 Has the original commitment been implemented?

☒ YES STOP. You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in CCEF summary report or in a docketed letter specific to the commitment change.

☐ NO STOP. Notify Site Regulatory Assurance. Timely notification of intended change to NRC is required if commitment change is pursued. Attach documentation of NRC notification.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ YES Go to Question 5.2.

☐ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ YES Revise the commitment and notify NRC of revised commitment in next CCEF summary report.

☐ NO Revise commitment. No NRC notification is required.

REFERENCES

List below the documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number / ID	Description
EN-DC-115	Engineering Change Process

Attachment 1 – List of River Bend Commitments to be Made Historic

LRS I.D.	Source Document
P-15747	RBG-32101
P-15748	RBG-35285
P-14125	RBG-23305
P-15524	RBG-35377
P-15532	RBG-39584
P-15533	RBG-41731 (RBF1-95-0177)
P-15746	RBG-29588

Commitment Descriptions

LRS I.D.	Source Document	
P-15747	RBG-32101	[CONCERNING "FAILURE TO FOLLOW PROCEDURE FOR MODIFICATION OF PERMANENT PLANT EQUIPMENT"] "ENG-3-006 WAS REVISED TO PROVIDE A MECHANISM FOR ENGINEERING TO EVALUATE AND AUTHORIZE FIELD WORK PARALLEL TO THE DEVELOPMENT OF THE MODIFICATION REQUEST."
P-15748	RBG-35285	[EVALUATION OF THE ENGINEERING/MAINTENANCE PLANNING RESPONSIBILITY AND INTERFACE] (REVISING)...THE MR PROCEDURE (ENG-3-006) TO REQUIRE THE DESIGN ENGINEER TO PROVIDE PRECAUTIONARY STATEMENTS IN THE MR TO ADVISE THE MAINTENANCE PLANNER OF POTENTIAL ESF ACTUATIONS.
P-14125	RBG-23305	[CONCERNING NON-SEISMICALLY QUALIFIED TUBING BEING INSTALLED IN THE PENETRATION VALVE LEAKAGE CONTROL SYSTEM (PVLCS)] THE DESIGN CONTROL PROCEDURE ENG-3-006 WAS REVISED TO REQUIRE PERFORMANCE OF A USQD PRIOR TO DESIGN APPROVAL.
P-15524	RBG-35377	THE REVISED MR PROCESS PROVIDES FOR TECHNICAL SPECIFICATION REVIEW EARLY IN THE MR (DESIGN) PROCESS. 09331 - TO ADDRESS TIMELINESS OF LICENSE AMENDMENT SUBMITTALS, THE REVISED MR PROCESS PROVIDES FOR TECHNICAL SPECIFICATION REVIEW EARLY IN THE MR (DESIGN) PROCESS.
P-15532	RBG-39584	10631 - ... THE DESIGN AND POST-DESIGN REVIEW PROCESSES WILL BE REVISED TO INSTRUCT THE ENGINEER TO LOOK AT ADVERSE IMPACTS TO PERIPHERAL SYSTEMS ...
P-15533	~~RBF1-0177~~ RBG-41731 (RBF1-95-0177)	REVISED ENG-3-033 TO ADDRESS THE RECOMMENDATION FOR CROSS-DISCIPLINARY REVIEWS FOR MULTI-DISCIPLINARY MODIFICATIONS. SPECIFIC QUESTIONS CONCERNING **PENETRATION PROTECTION** WILL BE INCLUDED IN THE "ELECTRICAL IMPACT PROGRAM REVIEW," PER EDP-AA-80 "MODIFICATION FORMS," AND THE "DESIGN INPUT GUIDE" PER EDP-AA-81 "DESIGN INPUTS."
P-15746	RBG-29588	ENG-3-006 HAS BEEN FURTHER REVISED TO REQUIRE THE COMPLETION OF A CANCELLATION CHECKLIST IF AN INDIVIDUAL WISHES TO CANCEL AN MR. THIS CHECKLIST REQUIRES DOCUMENTATION OF OTHER MECHANISMS FOR IMPLEMENTATION OF CORRECTIVE ACTIONS OR JUSTIFICATION AS TO WHY THE CORRECTIVE ACTION IS NO LONGER REQUIRED. THE SUPERVISOR'S AND THE RESPONSIBLE ENGINEER'S APPROVAL ARE REQUIRED FOR THIS NEW CHECKLIST.

LRS/CMS I.D. 17156 Site Reg. Assurance Tracking Number: NA

Source Document / Date: RBG-47472 August 2, 2012

Commitment:
- ☐ Cancellation
- ☒ Historic/Hist/Retired/NEI
- ☐ Revision
- ☐ Superseded

Has the original commitment been implemented? ☐ YES ☒ NO, stop and notify site Regulatory Assurance

Original Commitment Description:
Entergy will continue to investigate and evaluate for suitability alternative inspection methods, such as the remote camera suggested by the NRC, for the third and subsequent ISI intervals as long as the impracticality remains.

Note: referring to the examination of CRD welds.

Revised Commitment Description:
NA

Summary of Justification for Change (attach additional sheets if necessary):
See Attached EC Reply EC-72419

Prepared By: Cecil Glass / [signature] Date: 5/31/2017
Print Name / Signature

Management Approval: Daniel Durocher / [signature] Date: 5/31/2017
Print Name / Signature

Forward completed form to site Regulatory Assurance.

Reg. Assurance Management (or designee) Concurrence: Tim Schenk [signature] Date: 5-31-17
Print Name / Signature

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program Manual, Fire Protection Program, or Security Plan?

☐ YES STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54, 10 CFR 50.59, 10 CFR 50.55) to evaluate commitment

☒ NO Go to Part II.

PART II

2.1 Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its specified safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its specified safety function?

☐ YES Go to Question 2.2.

☒ NO Continue with Part III. Briefly describe rationale:

Improvements in technology are continually investigated to improve examination methods, quality and accessibility. Based on this and the discussion contained in EC-72419, changing Commitment No. 17156 from Active to Historical could not negatively impact the ability of a SSC to perform its safety function, or the ability of plant personnel to ensure the SSC is capable of performing its specified safety function.

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

- Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ YES ☐ NO Describe basis below

- Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ YES ☐ NO Describe basis below

- Does the revised commitment involve a significant reduction in a margin of safety?

☐ YES ☐ NO Describe basis below

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain necessary approvals. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required. IF all three questions are answered NO, THEN go to Part III. (Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an obligation (e.g., rule, regulation, order or license condition)?

☐ YES Go to Question 3.2.

☒ NO Continue with Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary). IF commitment not yet implemented, THEN notify NRC of revised commitment date prior to the original date in which the commitment was to be completed/satisfied.

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

☒ NO Continue with Part V.

4.2 Has the original commitment been implemented?

☐ YES STOP. You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in CCEF summary report or in a docketed letter specific to the commitment change.

☐ NO STOP. Notify Site Regulatory Assurance. Timely notification of intended change to NRC is required if commitment change is pursued. Attach documentation of NRC notification.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ YES Go to Question 5.2.

☒ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ YES Revise the commitment and notify NRC of revised commitment in next CCEF summary report.

☐ NO Revise commitment. No NRC notification is required.

REFERENCES

List below the documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number / ID	Description
See EC-72419	JUSTIFICATION TO RETIRE COMMITMENT 17156

The purpose of this EC reply is to provide justification for the Retirement of Commitment 17156 "Entergy will continue to investigate and evaluate for suitability alternative inspection methods, such as the remote camera suggested by the NRC, for the third and subsequent ISI intervals as long as the impracticality remains".

By letter dated August 3, 2011 (Ref. 1), as supplemented by letters dated April 16 (Ref. 3), and August 2, 2012 (Ref. 5), Entergy Operations, Inc. submitted Relief Requests (RR) RBS-ISI-016 and RBS-ISI-17 for the second 10-Year Inservice Inspection (ISI) Interval Program at the River Bend Station (RBS).

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the RRs RBS-ISI-016 and RBS-ISI-017 and concluded that compliance with the examination coverage requirements of Title 10 of the Code of Federal Regulations (10 CFR) Part 50 Section 50.55a(g)(5)(iii) for the ASME Class 1 Category B-O, Pressure Retaining Welds in Control Rod Housings and the ASME Class 2 pressure retaining welds in certain pumps and valves is impractical, thus fulfilling the technical requirements of 10 CFR 50.55a(g)(6)(i) (Ref. 6).

By letter dated August 2, 2012 (Ref. 5), with regards to RBS-ISI-016, RBS committed to "investigate and evaluate for suitability alternative inspection methods, such as the remote camera suggested by the NRC, for the third and subsequent ISI intervals as long as the impracticality remains." RBS also acknowledged that it would have to submit an alternative, if found possible to perform a visual examination by remote camera in lieu of the ASME Code requirements. *[RBS Commitment No. A-17156]*

Based on discussions with the NSSS Supplier, and investigation with respect to use of the remote visual devices (camera), a suitable remote camera that could overcome the space allowed between the control rod drive housings and the incore housing could not be located. Therefore, an enhanced visual technique was not used during the recent refueling outage (RF19), which concluded in March of 2017.

ASME Section XI Subsection IWB (IWB-2500-1), Item B14.10 Examination Category B-O requires 10 % of the Parts to be examined. There are thirty six (36) Peripheral CRD Housings, four (4) CRD Housings were selected to be examined during RF19 for weld No. 2. The following welds were examined using the liquid penetrant surface examination technique.

COMPONENT NO	CATEGORY	ITEM NUMBER	DESCRIPTION
B13-D008-04/21-WELD-2	B-O	B14.10	Welds in CRD Housing
B13-D008-04/25-WELD-2	B-O	B14.10	Welds in CRD Housing
B13-D008-04/29-WELD-2	B-O	B14.10	Welds in CRD Housing
B13-0008-04/33-WELD-2	B-O	B14.10	Welds in CRD Housing

The liquid penetrant surface examination for the above welds did not identify any issues or concerns, i.e. the examination results were deemed acceptable. Additionally, as stated via reference 1 for RR RBS-ISI-016, the subject welds received a VT-2 examination with the Reactor Coolant Pressure Boundary system leakage test conducted prior to startup from each refueling outage. Under-Vessel Drywell leakage was also monitored during this operational cycle.

With respect to CRD Housing tube-to-flange Weld # 1, the NRC staff has previously concluded via Reference 6, that there was reasonable assurance of the structural and leak tight integrity of the welds under consideration may be obtained through the use of the licensee's proposed alternative (i.e., VT-2 examinations).

Improvements in technology are continually investigated to improve examination methods, quality and accessibility. Based on this and the above discussion, retaining the commitment to further investigate and research for a remote visual device are not warranted to enhance examination coverage of the ASME Class 1 Category B-O, Pressure Retaining Welds in Control Rod Housings and the ASME Class 2 pressure retaining welds in certain pumps and valves.

REFERENCES:

1. Entergy Letter to NRC dated August 03, 2011, Requests for Relief RBS-ISI-016 and RBS-ISI-017, Requests for Relief from ASME Code Section XI Inservice Inspection Requirements for Pressure Retaining Welds in Control Rod Housings and Pressure Retaining Welds in Pumps and Valves (ML11221A164/RBG-47166).

2. NRC Email dated February 1, 2012, River Bend Station Request for Additional Information Regarding RR RBS-ISI-016 and RBS-ISI-017 (ML120320214).

3. Entergy Letter to NRC dated April 16, 2012, Supplement to Request for Relief RBS-ISI-016 and RBS-ISI-017 (ML12110A409/RBG-47233).

4. NRC Email dated July 12, 2012, River Bend Station Request for Additional Information Regarding Relief Requests RBS-ISI-016 and RBS-ISI-017 (ML120320214).

5. Entergy Letter to NRC dated August 2, 2012, Supplement to RBS-ISI-016 and RBS-ISI-017 Requests for Relief from ASME Code Section XI Inservice Inspection Requirements for Pressure Retaining Welds in Control Rod Housings and Pressure Retaining Welds in Pumps and Valves (ML12234A399/RBG-47272).

6. NRC Safety Evaluation Report dated August 31, 2012 River Bend Station, Unit 1 -RBS-ISI-016 and RBS-ISI-017 Proposed Alternative to 10 CFR 50.55a Examination Requirements for Reactor Pressure Vessel Weld Inspections (TAC NOS. ME6845 and ME6844) (ML12235A308)

RBG-47776

ATTACHMENT 4

RBS USAR REVISION 25 REVISED PAGES LIST

USAR Revision 25 Affected Pages

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9A.2-10	09A.02-057
9A.2-78	09A.02-057
9B.4-21	09A.02-057
7.7-37a	07.07-059
7.7-38	07.07-059
6.3-5	06.03-037
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7.3-9	07.03-188
9.3-6a	09.03-396
9a.2-74	09A.02-058
9a.2-76	09A.02-058
9.2-59	09.02-401
7-vii	09.02-401
12.4-6	12.04-002
8.3-52	13.01-069
9A.3-2	13.01-069
9B.4-7	13.01-069
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13-ii	13.01-069
13-iii	13.01-069
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13.2-12	13.01-069
13.2-13	13.01-069
13.4-4	13.01-069
13.5-19	13.01-069
13.5-20	13.01-069
13.5-21	13.01-069
9.4-11	07.03-205
9.2-59	09.02-059
7-vii	09.02-404
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12.2-7	11.04-020
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11-iv	11.04-020
12-ii	11.04-020
15.2-16	15.02-009
15.2-18	15.02-009
15.2-19	15.02-009
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11-iv	11.04-019
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9.2-28	09.02-411
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9.1-76	09.01-106
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9A.3-31	09.05-189
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8.3-87	08.03-145
8.3-89	08.03-145
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6-xxa	15B.00-021
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6.3-34	15B.00-021
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9.2-11	09.02-422
5.2-10	15.02-010
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15.2-20	15.02-010
1.4-1	01.04-004
13.1-8	01.04-004
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9.2-62	09.02-405
6.4-3	06.04-012
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7.7-28a	10.02-020
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10.2-3	10.02-020
10.2-4	10.02-020
10.2-5	10.02-020
10.2-11	10.02-020
10.2-11a	10.02-020
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15.1-9	10.02-020
15.1-10	10.02-020
15.1-11	10.02-020
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15.8-2	10.02-020
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1.8-1, sh 192	01.08-063
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8.3-7, sh 2	09.02-401
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1.8-1, sh 202	11.02.074
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9.2-11	09.02-417
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9A.2-36	09A.02-060
9A.2-37	09A.02-060
6.2-40 sh 1	06.02-137
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15.0-1 sh 4	10.02-020
15.0-1a sh 1	10.02-020
15.1-1a sh 5	10.02-020
15.8-4	10.02-020
15.1-4	10.02-020
15A.6-2 sh 2	10.02-020
1.8-1 sh 222	13.02-019
1.8-1, sh 237	13.02-019
3.6A-30b sh 1	03.06A-012
3.6A-30b sh 2	03.06A-012
3.6A-30b sh 3	03.06A-012
3.6A-30b sh 4	03.06A-012
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7.3-14, sh 4	09.02-401
7.3-14, sh 5	09.02-401
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7.3-14, sh 9	09.02-401
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7.3-14, sh 14	09.02-401
7.3-14, sh 15	09.02-401
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7.3-14, sh 19	09.02-401
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13.2-1, sh 2	13.01-069
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7.3-23 sh 42	09.02-059
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9.2-8j	09.02-059
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6.2-65	06.02-133
7.3-14 sh 1	09.02-404
7.3-14 sh 2	09.02-404
7.3-14 sh 3	09.02-404
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7.3-14 sh 12	09.02-404
7.3-14 sh 13	09.02-404
7.3-14 sh 14	09.02-404
7.3-14 sh 15	09.02-404
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7.3-14 sh 22	09.02-404
7.3-14 sh 23	09.02-404

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7.3-14 sh 19	07.03-217
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10.4-6B	10.04-244
10.4-6B	10.04-246
10.4-6b	10.04-245
9.5-1b	09.05-192
9.2-1d	09.02-406
9.2-1e	09.02-406
9.3-1e	09.02-406
9.3-8c	09.03-404
9.2-21c	09.02-416
9.2-8j	09.02-418
7.3-13 sh 1	07.03-211
10.4-3b	10.04-249
10.4-3b	11.02.074
11.2-1g	11.02.074
6.3-11a	15B.00-021
6.3-12a	15B.00-021
6.3-13a	15B.00-021
6.3-14a	15B.00-021
6.3-15a	15B.00-021
6.3-16a	15B.00-021
6.3-17a	15B.00-021
6.3-18a	15B.00-021
9.2-1F	09.02-409
10.4-3b	10.04-251
7.3-14, SH 3	07.03-213
9.3-16C	09.03-405
6.2-73A	06.02-134
9.2-1H	09.02-420
9.4-1a	09.04-194
9.2-1c	09.02-421

Affected Figure	LCNNUM
9.2-1e	09.02-421
9.2-1f	09.02-421
7.3-7, SH 1	07.03-218
7.3-7, SH 1	07.03-219
5.4-12c	05.04-197
7.6-8, sh 7	07.06-053
15.2-6	15.02-010
5.4-12c	05.04-199
9.3-7b	09.03-407
9.3-1c	09.03-406
9.3-7b	09.03-408
5.4-12c	05.04-200
9.2-1b	09.02-424
9.5-2D	09.05-193
7.3-18 sh 2	07.03-225
7.3-9 SH 5	07.03-221
7.3-20 SH 5	07.03-222
7.3-18 SH 2	07.03-224
7.3-20 sh 12	07.03-223
7.3-9 sh 1	07.03-220
5.1-3A	05.01-018
9.5-2d	09.05-196
9.5-1e	09.05-195
9.2-11	09.02-417
9.2-22	09.02-417
9.2-23	09.02-417
1.2-44	09.02-417
6.2-73a	06.02-175
6.2-73b	06.02-175
6.2-73a	06.02-135
6.2-73b	06.02-135
9.5-2d	09.05-197

Affected Figure	LCNNUM
9.5-2b	09.05-171
9.3-7l	09.03-410
9.3-4b	09.03-410
9.3-7m	09.03-411
9.2-8h	09.02-419
7.3-22, sh 7	07.03-227
9.2-1F	09.02-414
5.4-12c	09.03-412
9.3-24a	09.03-412
9.5-2d	09.05-205
9.2-1d	09.02-425
15.1-4	10.02-020
10.4-7d	10.02-020
10.4-7k	10.02-020
10.4-7c	10.02-020
10.4-6a	10.02-020
10.3-1c	10.02-020
9.2-1f	09.02-408
7.3-14, SH 22	07.03-230
7.3-12 sh 1	07.03-228
7.3-13 sh 1	07.03-228
3.6A-19	03.06A-012
9.4-6a	09.04-195
9.5-1c	09.05-198
8.1-6	08.01-028
13.1-2	13.01-070
13.1-7	13.01-070
7.3-12, sh 2	07.03-231
7.3-13, sh 2	07.03-231
7.3-13, sh 2a	07.03-231
9.3-12f	11.02-078
11.2-1f	11.02-078

Affected Figure	LCNNUM
11.2-1j	11.02-078
9.2-1a	09.02-427
9.5-1a	09.05-199
9.5-1A	09.05-200
9.5-1A	09.05-201
9.5-1A	09.05-202
9.5-1A	09.05-203
7.3-22 sh 6	07.03-226
9.2-3b	09.02-429
9.2-1D	09.02-430