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10 CFR 50.59 (d)(2)

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Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Report of Changes, Tests and Experiments

Pursuant to the requirements of 10 CFR 50.59, "Changes, tests and experiments," paragraph (d)(2), Byron Station is providing the required report for Facility Operating License Nos. NPF-37 and NPF-66. This report is provided for the 2005 and 2006 calendar years and consists of 50.59 Review Coversheets for changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR) and tests or experiments not described in the UFSAR.

This report also includes 50.59 Review Coversheets for changes not previously submitted for the 2004 calendar year. These 50.59 Review Coversheets are associated with changes, tests and experiments not yet completed at the time of the last report.

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

Respectfully,

David M. Hoots
Site Vice President
Byron Nuclear Generating Station

DMH/RC/rah

Attachment 1, Byron Station 10 CFR 50.59 Report

Attachment 1

Byron Station 10 CFR 50.59 Report

10 CFR 50.59 Review Coversheets for Calendar Years 2005 and 2006

and

10 CFR 50.59 Review Coversheets not previously submitted for Calendar Year 2004

Design Change Packages (DCP), Drawing Change Requests (DCR), Engineering Changes (EC), and Temporary Modifications (TMOD)	
1.	6G-04-0003 Rev. 0
2.	6G-04-0005 Rev. 0
3.	6G-04-0006 Rev. 1
4.	6G-04-0007 Rev. 0
5.	6G-06-0008 Rev. 0
6.	6G-05-0001 Rev. 0
7.	6G-06-0003 Rev. 0

50.59 REVIEW COVERSHEET FORM

Station: Byron Unit1

Activity/Document Number: EC #343598 **Revision Number:** 0

Title: DEH Replacement

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

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

EC # 343598 replaces the existing Westinghouse Mod II Digital Electro-Hydraulic (DEH) control system for the main turbine with a redundant fault tolerant Westinghouse Ovation based Distributed Control System (DCS). All control algorithms and processes are redundant and configured to allow unrestricted operation with a single failure, and to facilitate on-line replacement of a failed component. The DCS has a continuous self-diagnostics that will trigger an alarm if a problem has been detected. The primary function of the new turbine control system (TCS) is to control the turbine in maintaining proper speed and load in the same manner as provided by the existing DEH system. In addition to the original DEH control functions, added Turbine Control System (TCS) functions include turbine protection against hazardous conditions, Turbine and Generator Temperature Monitoring (TGTMS), and Moisture Separator Reheater (MSR) controls. The modification will eliminate the current mechanical overspeed trip and replace it with a new electronic trip design. The modification will also change how the DEH responds to grid underfrequency events.

The existing hard control stations will be replaced by new operator workstations in the main control room and an engineering work station/historian in the computer room. The new system controls based on soft graphical interfaces will provide the same functionality as the existing equipment so that the operators can operate the TCS in a safe, predictable, and controlled manner.

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

The existing Digital Electro-Hydraulic (DEH) system has very little redundancy and fault tolerance. Due to the inherent limitations of the technology and design, numerous single failure points exist that could trip the turbine or initiate a plant transient. The existing DEH system requires extensive setup and calibration each refueling outage. The existing DEH has limited diagnostic capability, which has resulted in unit trips due to lack of identification of failing components.

The new DCS includes extensive design features that provide the operator with information concerning the status of the new system. These include computer generated graphical interface screens, system failure alarms and indications, the ability to save for further review historical data and system information. The displays, alarms and event logging information that is available from the new system is more advanced and useful to the operator for detecting the cause of a transient. In addition, the DCS provides a single point of interface for the operator to perform functions associated with the DEH system.

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

The new DCS system functions are modeled based on the existing system and includes an improved control strategy. The new system does not involve the various sensors and associated setpoints that are used by the Reactor

Station: Byron Unit1

Activity/Document Number: EC #343598 **Revision Number:** 0

Title: DEH Replacement

Protection System (RPS) to initiate a preventative reactor trip. The modification does not impact the actuators. Changes are being made to various system performance requirements to enhance performance and reliability by using redundant inputs and outputs. Therefore, the new DEH system will contain all the functions of the existing system with the following changes as noted below:

- Two redundant Ovation controllers per drop are provided with the TCS. Only one controller in each drop is in control at ant time. The other is monitoring conditions and ready for hot swap over in case of a diagnosed failure. Each of the drops provides turbine protection and can cause a turbine trip. Each drop will protect against turbine overspeed conditions and will include a hard-wired overspeed trip system that will eliminate the mechanical overspeed. The system is independent of the controller logic and failure of both controllers in one drop. The hard-wired speed sensing modules will respond to turbine overspeed conditions in less than 10 msec.

- The controller loop processing speeds for all speed and load control shall be set at 250 ms to ensure all dynamics within the system can be controlled reliably. This setting is based on Westinghouse experience with similar processes in other turbine protection applications. The control valve interfaces will operate at 10ms loop to be highly responsive to minimize error between controller setpoint and valve position.

- The Front Standard upgrade will include three new speed probes, which will be sensed by Ovation speed detector modules in the Emergency Trip System (ETS) controller. These speed modules have a independent trip subsystem built into them which are configured in a 2/3 configuration to trip the turbine via a new ETS trip manifold. The three control channel speed detector modules in the Operator Automatic (OA) / Overspeed Protection Control (OPC) controller will be configured also in a 2/3 configuration to trip the turbine via a second OA/OPC trip manifold. The additional OA/OPC controller provides for an independent and diverse means of overspeed protection. The manual trip from the main control board in the ETS and OP/OPC controller is processed through the controller and additionally independent of the controller for added protection.

- The controller loop processing speeds for all protection functions such as OPC, ETS, and overspeed shall be set at 50 ms to ensure that turbine protection is provided. This setting is based on Westinghouse experience with similar processes in other turbine protection applications.

- The existing hydraulic trip logic has been modified such that two electrical trip manifolds using a one out of two taken twice solenoid arrangement is installed to provide redundancy. Each trip manifold is assigned to one of the two TCS controllers and actuated only by that controller.

- The modulating valves such as throttle and governor valves, will switch to manual logic that is activated by detection of multiple bad sensors or multiple input channel bad quality (i.e. sensor signal out of range or failed). The switch of the associated controls from automatic to manual mode is provided in a bumpless manner to avoid causing system transients.

- Redundant outputs are provided for the governor valves. Discrepancy logic is included for each governor valve to detect and alarm invalid governor valve positions, demand and errors, between primary valve interface and the backup.

- Discrepancy logic is also provided to detect and alarm invalid valve position indications for reheat and intercept stop valves.

Station: Byron Unit1

Activity/Document Number: EC #343598 **Revision Number:** 0

Title: DEH Replacement

- When a controller detects a trip condition, it will send a trip signal to the other controller to cause it to also trip. These contacts are configured as "and" in the control logic to preclude false actuation in normal operating conditions.

- The single pressure switches for monitoring condenser vacuum, thrust bearing oil pressure and bearing oils pressure will be replaced with three transmitters for each application. The logic for each system will be changed to a 2 out of 3 logic to increase reliability.

- A local manual electrical trip switch has been added to the turbine front standard to replace manual trip lever that will cause a turbine trip via both trip manifolds when actuated. A manual emergency stop pushbutton is also available on the main control room panel independent of the digital system as in the existing design. The controls shall have a one out of one switch logic for manual tripping.

- The TGTMS will be implemented in Ovation controller (Drop 2/52). The TGTMS collects data from a number of temperature detectors located on the turbine generator and performs calculations that will be displayed on an Ovation Operator station in the control room. The TGTMS also has hardwired annunciator outputs to two main control room annunciators for "Warning Setpoint Exceeded" and "Trip Setpoint Exceeded".

- EC 348178 modified the DEHC turbine control software input parameter SPDBMINS to reduce the potential for reactor power transients because of the DEHC response to grid under frequency events. The new system eliminates the potential for this event by improved monitoring of frequency and factoring of reactor power level into the software.

- Removal of recorders 1TR-TS001B (LP Turbine Metal Temperature), 1TR-TS002A, B, C & D (Turbine Vibration, Expansion and Eccentricity, and 1ZR-TS003 (GV Valve Position, Rotor Speed, Rotor Position), and removal of LP turbine inlet steam temperature channels from other recorders 1TR-TS001D, E & F and rearranging the remaining recorders. The Ovation controller will replace the function of the removed recorders.

The electrical portion of the DEH will maintain the same functional and interface requirements as the current equipment with improved performance, except as follows:

a. Quantity, type and performance requirements for field and main control room instrumentation providing analog or digital input signals will be altered by this design change.

b. Quantity, type and performance requirements for field and main control room instrumentation driven by analog or digital output signals will be altered by this design change. Some analog output signals for recorders, indicators and computer inputs will be eliminated since these instruments will be driven directly by analog current signals. Some of digital output signals that were used for light indication or alarm functions in the existing system will be eliminated and replaced by new outputs performing different function since certain display and alarm functions in the new system will be replaced by new alarm and indication functions. Some digital outputs will be added to provide annunciation for Drop 2/52 emergency trip system controller and Drop 3/53 operator automatic and overspeed protection control function trouble alarms as a part of this EC.

Station: Byron Unit1

Activity/Document Number: EC #343598 **Revision Number:** 0

Title: DEH Replacement

c. 120 VAC power supply configuration has been altered by this design change. The new design provides redundant power supply for the new system using the existing power sources.

The new DCS system has been designed as a highly reliable system. This design is achieved through implementation of a combination of highly reliable components and fault tolerant design. The DCS self-tests and on-line diagnostics are capable of identifying failures in I/O cards, power supplies, processors and any of the modular components. These features identify the presence of a fault and determine the location of failure to a replaceable module level. Therefore, the DEH design assures that a single active component within the new system will neither result in a loss of continuous validated demand signals to TCS valves or operator displays nor will failure of individual modules cause the DCS system to trip the turbine or introduce a plant transient. The control elements in the new system are configured to allow unrestricted operation with a single failure, and to facilitate online replacement of a failed component. The existing control system is not-fault tolerant in automatic mode and does not provide on-line fault detection and failure response capability. Therefore, the new system improves the reliability of the entire DCS system.

Guidance was taken from EPRI Guidelines for Electromagnetic Interference Testing and for Licensing of Digital Upgrades (not specifically required for non-safety related upgrades) in the design of this change.

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

This modification replaces the DEH Turbine Controls system with a more reliable system that uses logic based on the existing plant design updated to current control logic standard design. The DEH control system performs two main functions: control of turbine speed and control of turbine load. The controller compares signals representing turbine speed and first-stage pressure with reference values initiated by a load demand signal. The controller then puts out a comparison signal which actuates hydraulic control of the main turbine governor valves to match generator output to load demand. The system being modified is non-safety related and is not credited with any accident mitigation functions. The modification does not impact any of the devices or associated setpoints that provide anticipatory reactor trips due to turbine operating events. The modification does not change any valve response time or controls of any turbine valves. The response time of the valves is not credited in any accident analyses. The DEH system is one of a number of possible initiators of a plant turbine trip event (UFSAR 15.2.3). A detailed failure modes and effects analyses prepared as part of the modification shows that there is no significant increase in frequency of this event caused by the modification. Analysis has also shown that the probability of a turbine overspeed event has been reduced. The installation of the modified system does not result in any adverse interactions with any important to safety equipment in the plant.

A full 50.59 evaluation was performed because of the replacement of the mechanical overspeed trip with a trip generated by an electronic device (UFSAR Change 10-066 & TRM Change 04-008) and the incorporation of digital design components with a modified logic.

There is no change to any procedures described in the UFSAR.

There is no change to the Operating License or Technical Specification required by this modification.

50.59 REVIEW COVERSHEET FORM

Station: Byron Unit1

Activity/Document Number: EC #343598 **Revision Number:** 0

Title: DEH Replacement

In conclusion, the proposed activity does require a 50.59 evaluation, and the evaluation concluded that NRC notification of the proposed activity is not required.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

	Applicability Review				
X	**50.59 Screening**	**50.59 Screening No.**	**6E-04-0088**	**Rev.**	**0**
X	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-04-0003**	**Rev.**	**0**

50.59 REVIEW COVERSHEET FORM

Station: Byron Units 1 and 2

Activity/Document Number: EC 340545 (U1) and EC 348121 (U2) **Revision Number:** 0

Title: Replace Steam Generator Blowdown Condenser Hotwell Pump Moter Controller

Description of Activity:

These changes will install replacement Steam Generator Blowdown Condenser Hotwell Pump variable speed drive Controllers (1/2SD01JA & 1/2SD01JB). The new controllers are Yaskawa/Magnetek Model GPD506 Digitally Controlled devices, and will replace the existing analog device controllers. The controllers provide power to the pumps identified as 1/2SD02PA & PB.

Both the existing and new controllers convert 480 VAC 3-phase power feeds to a 3-phase variable speed output to drive the condenser hotwell pumps at a speed based on a variable level signal to each controller. The new controllers utilize digital processing and digital technology for the power signal conversion. Obsolescence and expensive serviceability of existing internal circuit boards on the existing controllers necessitates a more economical state-of-the-art component. The new controllers utilize a microprocessor and associated software, firmware, and hardware to provide signal processing options.

Reason for Activity:

The intent of these changes is to improve the performance and reliability of the controllers and to reduce maintenance expenses.
The existing controller components are experiencing increased failure rates due to age degradation and are also becoming obsolete. Servicing and replacing available parts is not economical compared to installing a newer style controller. The most economical replacement is a commercially manufactured digital controller and has been recommended by Byron's servicing vendor.

Effect of Activity:

The purpose of the steam generator blowdown subsystem is to maintain the steam generator water chemistry within specified limits for optimum operation. The system is a subsystem of the Liquid Radwaste processing (WX) system. Continuous blowdown constantly removes impurities from the steam generators. The flow rate is varied as required to maintain the steam generator water chemistry within the required limits. During normal operation, blowdown is pumped from the steam generator blowdown condenser hotwells through the blowdown prefilters, the blowdown mixed-bed demineralizers, and the blowdown after filters to the condensate storage tanks or respective unit hotwell. In the event of high radioactive material in the purified effluent leaving the blowdown mixed-bed demineralizers, the effluent is diverted to the monitoring tanks.

These controller changes do not alter the system function nor any of the system flow paths, level controls, filtering processes, or component arrangements.

These changes affect the Steam Generator Blowdown Condenser Hotwell Pump Motor Controllers, which are components of the Steam Generator Blowdown (SD) system. The pumps are individually controlled and have a common suction and discharge path for parallel operation. The pumps respond to level changes in the hotwell condenser. Manual or automatic level control is selected from the operator station at the Radwaste Control Panel 0PL01J. Pump starts and pump parallel operation will not be affected differently using different controllers. The level sensing instrumentation will not be altered by these design changes.

Changes will also affect control wiring at Radwaste Control Panel 0PL01J. The design function of the pumps and the blowdown system will not be altered by these changes. The controllers interface with the Auxiliary Power (AP) and Annunciator (AN) systems. None of these systems will be affected by utilizing different controllers. The structural integrity of controller cabinet seismic supports has been evaluated and will not be adversely affected. No other plant

Station: Byron Units 1 and 2

Activity/Document Number: EC 340545 (U1) and EC 348121 (U2) **Revision Number:** 0

Title: Replace Steam Generator Blowdown Condenser Hotwell Pump Moter Controller

equipment or systems will be affected. No electrical design evaluations, load monitoring systems or calculations require revision.

The effect of the change will be the more reliable operation of the system with reduced maintenance costs.

Summary of Conclusion for the Activity's 50.59 Review:

Although this design change introduces no adverse effects to any component or system functions, the replacement of the motor controllers was conservatively treated as adverse due to being in the category of an analog to digital upgrade. Therefore, a 50.59 evaluation has been performed.

Although different components susceptible to failure are being installed, the result and consequences of a malfunction or failure are not different than previously analyzed in the UFSAR and are therefore bounded by existing UFSAR analyses. This change activity does not affect any accidents that have been evaluated in the UFSAR. The effects of minor secondary system pipe breaks are considered in the chapter 15.1 analysis of events causing Increase in Heat Removal by the Secondary System. A blowdown system piping break would be bounded by the analysis. The controller changeout is not affecting the pumps' integrity, the system piping integrity, nor any system boundaries. The motor controller function is to provide the variable speed changes to the pumps to control level in the hotwell condenser. Failure of the function would result in degradation of steam generator chemistry and adverse operational conditions, potentially leading to unit shutdown. This is the same result as with the existing controllers. In addition, based on a technical review of new components and construction, Engineering has determined that there will not be more than a minimal increase in the likelihood of a malfunction, initiation (frequency) of a transient, or the likelihood of an accident.

Although a malfunction or failure of the new controller could result in the loss of the associated pump, the loss of a pump is not an accident initiator. Controller failure would have no different result than failure of the original controller. Failure or loss of one controller/pump would not cause failure of the parallel pump. If a common-mode defect in the controllers occurred, the result may cause simultaneous loss of both pumps. However, the Steam Generator Blowdown Condenser Hotwell Pumps do not have a function that is credited in the UFSAR for accident mitigation. The loss of Steam Generator Blowdown function would eventually result in degraded steam generator chemistry, an operational concern with ultimate consequences being a manual shutdown of the unit. Manual unit shutdown is described in Chapter 15.0 as a Condition I event- Normal Operation and Operational Transients. Using digital devices for motor controllers in the blowdown system would not result in any different failures that would cause the failure to propagate to a more severe condition.

A malfunction of the new controller may result in the loss of a blowdown condenser hotwell pump or the inability of a blowdown condenser hotwell pump to produce the expected flow rate. These potential malfunctions are the same that exist with the original motor controllers. Controller malfunction, ranging from failure to power the motor to powering the motor to a trip condition, would lead to loss of the blowdown function. This situation is bounded by UFSAR analyses as an anticipated plant transient. However, the new controller is no more likely to malfunction than the existing control system. Microcomputers have been utilized for control applications in industrial complexes and power plants for many years.

Although failure of the new digital control system could be considered a new failure mechanism, the failure mode and effect remains the same as that of existing controller failures. Therefore, no new failure modes or accidents will be created, and the UFSAR analyses remains bounding.

50.59 REVIEW COVERSHEET FORM

Station: Byron Units 1 and 2

Activity/Document Number: EC 340545 (U1) and EC 348121 (U2) **Revision Number:** 0

Title: Replace Steam Generator Blowdown Condenser Hotwell Pump Moter Controller

The pump controllers affected by these changes do not provide any contribution as a fission product barrier, or affect any parameters which affect design basis limits for fission product barriers. The blowdown system piping and isolation valves at containment penetrations will not be affected by this activity. Therefore, these changes will not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

In summary, the conclusion of the evaluation was that installation of the new controllers does not require a License Amendment. Therefore, the proposed changes may be implemented per applicable procedures.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
X	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-04-0005**	**Rev.**	**0**

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC #337049 **Revision Number:** 1

101

Title: Installation of Temporary Pumps for Draining the 'A' SX Suction Piping

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

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

The EC installs temporary pumping systems to facilitate draining the "A" Essential Service Water (SX) System suction piping. The temporary pumps will take suction from the 1A and 2A SX Pump Strainer drains and pump the water to the common SX return header, 2SX03B-42", via existing drain valves 0SX241 and 0SX242. Two separate temporary pumping systems will be installed. Each system will contain a pump, suction piping and hose, discharge piping and hose, manual isolation valves, a discharge check valve at the tie in to the SX return header drains, and a temporary power supply. The temporary pumps, piping, manual valves, and hose will be non-safety related non-ASME components. The temporary pumps will be powered from a local 480V welding receptacle.

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

The SX suction piping to the 1A and 2A pumps needs to be drained to facilitate replacement of the suction isolation valves 1/2SX001A. Replacement of the valves during B1R13 requires entering a 144 hour LCOAR for the affected SX systems. Use of a temporary pump to drain the suction piping will minimize the time the system is out of service. Approximately 30,000 gallons of water need to be drained from the SX suction header. Returning the water to the SX system minimizes the amount of effluent released from the plant.

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

Installation of the temporary pumping systems will have no adverse effect on plant equipment. Prior to being put in service the temporary pumps and piping will be isolated from the operating SX system by closed manual isolation valves. The temporary equipment, piping, and hose will be located/routed such that it does not interact with any safety related plant equipment. The additional temporary loads on the permanent SX system piping have been evaluated and is acceptable.

When the temporary pumping systems are put in service approximately 125 gpm of SX water will be pumped by each temporary pump from the strainer drain to the return header. The SX strainer will be isolated from the operating SX system which will allow the temporary pump to transfer water from the SX pump suction to the common return header. The temporary pumps will be tested with the Essential Service Water Cooling Tower (SXCT) suction isolation valve (0SX138A) open. During testing, the temporary pumps will bypass a small amount of flow from the SX suction to the return header. When the SXCT suction isolation valve is closed, the temporary pumps will be used to power drain the suction piping. During the power drain operation, the drained water will increase the amount of water returned to the SXCT basins. As basin level increases the automatic basin level control system will respond by reducing normal makeup to the SXCTs. The power drain operation will have no affect on the operating SX train. The SX trains are normally operated crosstied, thus all SX supplied equipment will be cooled by the operating SX train.

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

The additional load of the temporary piping and hose on the permanent SX system piping has been evaluated and the stress remains within design allowables. Even though the pressure boundary components used for the temporary pumping system are not certified to ASME Section III requirements, the equipment and components are designed, manufactured, and tested for maximum working pressures above the design pressure of the SX system and the maximum pressure of the temporary pumps. The pressure retaining components for the temporary change are judged to be equivalent to the current piping and components in the SX cooling system. Thus the probability of a SX system moderate energy line break while the temporary piping system forms part of the SX system pressure boundary is judged not to be more than a minimal increase. The installation of the

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

Activity/Document Number: EC #337049 **Revision Number:** 1 101

Title: Installation of Temporary Pumps for Draining the 'A' SX Suction Piping

temporary pumping system will have no adverse impact on the operation of the SX system or any SX system components. The temporary equipment, piping, and hose will be located/routed such that it does not interact with any safety related plant equipment.

Any debris that would bypass the strainer will be transported to the 42" diameter SX return header and back out to the SXCTs, thus there is no increase in the likelihood of debris blocking a SX system heat exchanger or isolation valve. Any debris that can pass through the temporary pump would pass through the SXCT spray nozzles. The small amount of bypass flow (250 gpm) will have negligible impact on the SX system flow balance because the flow rate is very small compared to the normal flow through the SX system of approximately 24,000 gpm per unit. Thus there is no increase in the likelihood of a malfunction of an operating SX train while the temporary pumps are in operation. 101

The reduction in normal makeup flow to the SXCTs will not increase the likelihood of a malfunction of the SX makeup systems because the system is designed to automatically adjust for changing basin level in response to changes in evaporation, drift, system leakage, and/ or blowdown. Based on discussions with Chemistry, the temporary reduction in makeup flow will have negligible impact on the water chemistry of the SX system and will have no effect on the heat transfer rate of the system heat exchangers.

When the temporary piping system forms part of the SX system pressure boundary, a line break in the temporary piping would be terminated by shutdown of the temporary pump, manual isolation of the temporary suction piping isolation valves, and auto closure of a check valve located at the temporary piping tie-in to the return header. The suction piping is connected to the out of service SX suction header that is being drained, thus the loss of inventory does not adversely affect SX system operation. The temporary check valves located at the return header connection will prevent loss of inventory from the operating SX systems. A review of the flooding calculation shows that the flooding from a postulated break of the temporary piping system is bounded by postulated flooding from breaks of existing system piping. No safety related equipment other than the out of service "A" train SX equipment is located near the path of the temporary piping. Thus, a break of the temporary piping system will not result in spray or pipe whip that could adversely affect safe shutdown.

A hydraulic analysis of flow through the open safety related piping and gate valve shows that the break flow from the SX return header will be limited to ~1100 gpm for a postulated line break and failure of the check valve connected to the SX return header or a break in the non-safety piping between the check valve and isolation valve. Assuming the check valve fails, operator action would be taken to manually close the gate valve to terminate the event. The loss of SX inventory from the operating return header would not adversely affect SX system operation because the SXCT basin contains adequate inventory margin for safe shutdown or accident mitigation.

The temporary pumping system does not introduce the possibility of a new accident because the temporary installation is not an initiator of any accident. Failure of the temporary pump to function only affects the rate of drain down of the isolated SX suction header. Pressure boundary failures have been previously evaluated and remain bounding. A comparison of UFSAR identified failures indicates that the results of the failure modes resulting from this EC are bounded by those presented in the UFSAR.

The proposed change does not affect the fuel clad, Reactor Coolant System (RCS) pressure boundary, or containment integrity. SX system design functions are maintained to ensure that fission product barriers are not compromised due to a lack of safety related heat removal capability.

The additional load of the temporary piping and hose on the permanent SX system piping has been evaluated using the same method of evaluation (PIPSYS) described in the UFSAR. The evaluation methods for flooding associated with a moderate energy line break, SX basin water temperature, and SX makeup and minimum basin levels are not changed by the proposed activity.

Based upon the results of this evaluation, the activity may be implemented per plant procedures without obtaining a License Amendment.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: EC #337049 **Revision Number:** 1 101

Title: Installation of Temporary Pumps for Draining the 'A' SX Suction Piping

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
X	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-04-0006	**Rev.**	1 101

50.59 REVIEW COVERSHEET FORM

Station: Byron Unit 2

Activity/Document Number: EC #343599 **Revision Number:** 0

Title: DEH Replacement

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

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

EC # 343599 replaces the existing Westinghouse Mod II Digital Electro-Hydraulic (DEH) control system for the main turbine with a redundant fault tolerant Westinghouse Ovation based Distributed Control System (DCS). All control algorithms and processes are redundant and configured to allow unrestricted operation with a single failure, and to facilitate on-line replacement of a failed component. The DCS has a continuous self-diagnostics that will trigger an alarm if a problem has been detected. The primary function of the new turbine control system (TCS) is to control the turbine in maintaining proper speed and load in the same manner as provided by the existing DEH system. In addition to the original DEH control functions, added Turbine Control System (TCS) functions include turbine protection against hazardous conditions, Turbine and Generator Temperature Monitoring (TGTMS), and Moisture Separator Reheater (MSR) controls. The modification will eliminate the current mechanical overspeed trip and replace it with a new electronic trip design. The modification will also change how the DEH responds to grid underfrequency events.

The existing hard control stations will be replaced by new operator workstations in the main control room and an engineering work station/historian in the computer room. The new system controls based on soft graphical interfaces will provide the same functionality as the existing equipment so that the operators can operate the TCS in a safe, predictable, and controlled manner.

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

The existing Digital Electro-Hydraulic (DEH) system has very little redundancy and fault tolerance. Due to the inherent limitations of the technology and design, numerous single failure points exist that could trip the turbine or initiate a plant transient. The existing DEH system requires extensive setup and calibration each refueling outage. The existing DEH has limited diagnostic capability, which has resulted in unit trips due to lack of identification of failing components.

The new DCS includes extensive design features that provide the operator with information concerning the status of the new system. These include computer generated graphical interface screens, system failure alarms and indications, the ability to save for further review historical data and system information. The displays, alarms and event logging information that is available from the new system is more advanced and useful to the operator for detecting the cause of a transient. In addition, the DCS provides a single point of interface for the operator to perform functions associated with the DEH system.

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

The new DCS system functions are modeled based on the existing system and includes an improved control strategy. The new system does not involve the various sensors and associated setpoints that are used by the Reactor Protection System (RPS) to initiate a preventative reactor trip. The modification does not impact the actuators.

50.59 REVIEW COVERSHEET FORM

Station: Byron Unit 2

Activity/Document Number: EC #343599 **Revision Number:** 0

Title: DEH Replacement

Changes are being made to various system performance requirements to enhance performance and reliability by using redundant inputs and outputs. Therefore, the new DEH system will contain all the functions of the existing system with the following changes as noted below:

- Two redundant Ovation controllers per drop are provided with the TCS. Only one controller in each drop is in control at any time. The other is monitoring conditions and ready for hot swap over in case of a diagnosed failure. Each of the drops provides turbine protection and can cause a turbine trip. Each drop will protect against turbine overspeed conditions and will include a hard-wired overspeed trip system that will eliminate the mechanical overspeed. The system is independent of the controller logic and failure of both controllers in one drop. The hard-wired speed sensing modules will respond to turbine overspeed conditions in less than 10 msec.

- The controller loop processing speeds for all speed and load control shall be set at 250 msec to ensure all dynamics within the system can be controlled reliably. This setting is based on Westinghouse experience with similar processes in other turbine protection applications. The control valve interfaces will operate at 10ms loop to be highly responsive to minimize error between controller setpoint and valve position.

- The Front Standard upgrade will include three new speed probes, which will be sensed by Ovation speed detector modules in the Emergency Trip System (ETS) controller. These speed modules have a independent trip subsystem built into them which are configured in a 2/3 configuration to trip the turbine via a new ETS trip manifold. The three control channel speed detector modules in the Operator Automatic (OA) / Overspeed Protection Control (OPC) controller will be configured also in a 2/3 configuration to trip the turbine via a second OA/OPC trip manifold. The additional OA/OPC controller provides for an independent and diverse means of overspeed protection. The manual trip from the main control board in the ETS and OP/OPC controller is processed through the controller and additionally independent of the controller for added protection.

- The controller loop processing speeds for all protection functions such as OPC, ETS, and overspeed shall be set at 50 ms to ensure that turbine protection is provided. This setting is based on Westinghouse experience with similar processes in other turbine protection applications.

- The existing hydraulic trip logic has been modified such that two electrical trip manifolds using a one out of two taken twice solenoid arrangement is installed to provide redundancy. Each trip manifold is assigned to one of the two TCS controllers and actuated only by that controller.

- The modulating valves such as throttle and governor valves, will switch to manual logic that is activated by detection of multiple bad sensors or multiple input channel bad quality (i.e. sensor signal out of range or failed). The switch of the associated controls from automatic to manual mode is provided in a bumpless manner to avoid causing system transients.

- Redundant outputs are provided for the governor valves. Discrepancy logic is included for each governor valve to detect and alarm invalid governor valve positions, demand and errors, between primary valve interface and the backup.

- Discrepancy logic is also provided to detect and alarm invalid valve position indications for reheat and intercept stop valves.

Station: Byron Unit 2

Activity/Document Number: EC #343599 **Revision Number:** 0

Title: DEH Replacement

- When a controller detects a trip condition, it will send a trip signal to the other controller to cause it to also trip. These contacts are configured as "and" in the control logic to preclude false actuation in normal operating conditions.

- The single pressure switches for monitoring condenser vacuum, thrust bearing oil pressure, and bearing oil pressure will be replaced with three transmitters for each application. The logic for each system will be changed to a 2 out of 3 logic to increase reliability.

- A local manual electrical trip switch has been added to the turbine front standard to replace manual trip lever that will cause a turbine trip via both trip manifolds when actuated. A manual emergency stop pushbutton is also available on the main control room panel independent of the digital system as in the existing design. The controls shall have a one out of one switch logic for manual tripping.

- The TGTMS will be implemented in Ovation controller (Drop 2/52). The TGTMS collects data from a number of temperature detectors located on the turbine generator and performs calculations that will be displayed on an Ovation Operator station in the control room. The TGTMS also has hardwired annunciator outputs to two main control room annunciators for "Warning Setpoint Exceeded" and "Trip Setpoint Exceeded".

- EC 348178 modified the DEHC turbine control software input parameter SPDBMINS to reduce the potential for reactor power transients because of the DEHC response to grid under frequency events. The new system eliminates the potential for this event by improved monitoring of frequency and factoring of reactor power level into the software.

- Removal of recorders 2TR-TS001B (LP Turbine Metal Temperature), 2VR-TS002A, B, C & D (Turbine Vibration, Expansion and Eccentricity, and 2ZR-TS003 (GV Valve Position, Rotor Speed, Rotor Position), and removal of LP turbine inlet steam temperature channels from other recorders 2TR-TS001D, E & F and rearranging the remaining recorders. The Ovation controller will replace the function of the removed recorders.

The electrical portion of the DEH will maintain the same functional and interface requirements as the current equipment with improved performance, except as follows:

a. Quantity, type and performance requirements for field and main control room instrumentation providing analog or digital input signals will be altered by this design change.

b. Quantity, type and performance requirements for field and main control room instrumentation driven by analog or digital output signals will be altered by this design change. Some analog output signals for recorders, indicators and computer inputs will be eliminated since these instruments will be driven directly by analog current signals. Some of digital output signals that were used for light indication or alarm functions in the existing system will be eliminated and replaced by new outputs performing different function since certain display and alarm functions in the new system will be replaced by new alarm and indication functions. Some digital outputs will be added to provide annunciation for Drop 2/52 emergency trip system controller and Drop 3/53 operator automatic and overspeed protection control function trouble alarms as a part of this EC.

c. 120 VAC power supply configuration has been altered by this design change. The new design provides redundant power supply for the new system using the existing power sources.

50.59 REVIEW COVERSHEET FORM

Station: Byron Unit 2

Activity/Document Number: EC #343599 **Revision Number:** 0

Title: DEH Replacement

The new DCS system has been designed as a highly reliable system. This design is achieved through implementation of a combination of highly reliable components and fault tolerant design. The DCS self-tests and on-line diagnostics are capable of identifying failures in I/O cards, power supplies, processors and any of the modular components. These features identify the presence of a fault and determine the location of failure to a replaceable module level. Therefore, the DEH design assures that a single active component within the new system will neither result in a loss of continuous validated demand signals to TCS valves or operator displays nor will failure of individual modules cause the DCS system to trip the turbine or introduce a plant transient. The control elements in the new system are configured to allow unrestricted operation with a single failure, and to facilitate online replacement of a failed component. The existing control system is not-fault tolerant in automatic mode and does not provide on-line fault detection and failure response capability. Therefore, the new system improves the reliability of the entire DCS system.

Guidance was taken from EPRI Guidelines for Electromagnetic Interference Testing and for Licensing of Digital Upgrades (not specifically required for non-safety related upgrades) in the design of this change.

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

This modification replaces the DEH Turbine Controls system with a more reliable system that uses logic based on the existing plant design updated to current control logic standard design. The DEH control system performs two main functions: control of turbine speed and control of turbine load. The controller compares signals representing turbine speed and first-stage pressure with reference values initiated by a load demand signal. The controller then puts out a comparison signal, which actuates hydraulic control of the main turbine governor valves to match generator output to load demand. The system being modified is non-safety related and is not credited with any accident mitigation functions. The modification does not impact any of the devices or associated setpoints that provide anticipatory reactor trips due to turbine operating events. The modification does not change any valve response time or controls of any turbine valves. The response time of the valves is not credited in any accident analyses. The DEH system is one of a number of possible initiators of a plant turbine trip event (UFSAR 15.2.3). A detailed failure modes and effects analyses prepared as part of the modification shows that there is no significant increase in frequency of this event caused by the modification. Analysis has also shown that the probability of a turbine overspeed event has been reduced. The installation of the modified system does not result in any adverse interactions with any important to safety equipment in the plant.

A full 50.59 evaluation was performed because of the replacement of the mechanical overspeed trip with a trip generated by an electronic device (UFSAR Change 10-066 & TRM Change 04-008) and the incorporation of digital design components with a modified logic.

There is no change to any procedures described in the UFSAR.

There is no change to the Operating License or Technical Specification required by this modification.

50.59 REVIEW COVERSHEET FORM

Station: Byron Unit 2

Activity/Document Number: EC #343599 **Revision Number:** 0

Title: DEH Replacement

In conclusion, the proposed activity does require a 50.59 evaluation, and the evaluation concluded that NRC notification of the proposed activity is not required.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

	Applicability Review				
X	**50.59 Screening**	**50.59 Screening No.**	**6E-04-0155**	**Rev.**	**0**
X	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-04-0007**	**Rev.**	**0**

50.59 REVIEW COVERSHEET FORM

Station: Byron Unit 0

Activity/Document Number: EC 349308 **Revision Number:** 0

Title: Upgrade to Spent Fuel Pool Bridge Crane 0FH05G Control System; remove position indexing software to free computer memory.

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

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

This activity will install revised software in the microprocessor-based Control System, located in the Control Cabinet of the non-safety related Spent Fuel Pool Bridge Crane (SFPBC) 0FH05G. The existing SFPBC application reads from and writes to an external status file that tracks the actual physical location of the crane. The data in this file is used to determine the location of the SFPBC and calculate the rack and cell location, which is then displayed on the operator's pendant. The SFPBC rack and cell location digital display on the pendants is for reference only; it is not required to be operational for fuel movement. Correct fuel assembly placement is the responsibility of the Fuel Handler and cognizant management personnel. The revised software will remove the position information displayed on the operator's pendant. Removal of this portion of the application software will also disable the bridge slow zone at the end of bridge travel and the interlock that prevents diagonal motion while the SFPBC is in the transfer canal. This activity would not change the equipment the operator is using. The same skills of the operator will be used to avoid/prevent errors. The Operator's control of the SFPBC and emergency shutdown of the SFPBC will not be changed. This activity does not require any hardware change.

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

This activity is expected to eliminate frequent failures of the microprocessor. Byron has experienced continuing problems with the Spent Fuel Pool Bridge Crane (SFPBC). Multiple Condition Reports have noted problems with the SFPBC controls. OPEX OE10936 - Spent Fuel Pool Bridge Crane Malfunction was issued to document these problems. In the crane control system, an error occasionally occurs when the program writes data to its solid-state disk drive. A data file is periodically updated by the position tracking system to store crane position information. Currently the control program is stored in the same physical solid-state disk drive as the data file that the program writes to; this results in not only a data storage failure but also loss of the operating program for the crane. This condition results in the frequent faults, causing the microprocessor to reboot or stall. In most cases the computer reboots and operations continue normally; infrequently the system fails to reboot and requires the motherboard to be removed and reprogrammed. While similar, Braidwood's hardware and software is different. Braidwood does not have an indexing system with their version of the equipment.

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

This activity installs revised software in the SFPBC control system. The revised software will remove the position indicating system. This activity also verifies that software associated with operational limits will not be changed. This change will eliminate the need to update the position tracking file and therefore should eliminate the periodic corruption of the associated tracking file and application program.

This activity would not change the equipment the operator is using. It removes the position indication. This activity will allow operation of the SFPBC to continue to run within parameters allowed by the Westinghouse Fuel Specification Instructions, Cautions, and Limitations for Handling New and Partially Spent Fuel Assemblies; F-5 Rev 17 dated 6-7-00. The same skills of the operator will be used to avoid/prevent errors. The Operator's control of the SFPBC and emergency shutdown of the SFPBC will not be changed.

50.59 REVIEW COVERSHEET FORM

Station: Byron Unit 0

Activity/Document Number: EC 349308 **Revision Number:** 0

Title: Upgrade to Spent Fuel Pool Bridge Crane 0FH05G Control System; remove position indexing software to free computer memory.

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

The non-safety related Spent Fuel Pool Bridge Crane (SFPBC) 0FH05G is used for handling fuel assemblies within the spent fuel storage area. This activity will install revised software, which will remove the position information displayed on the operator's pendant. The SFPBC rack and cell location digital display on the pendants is for reference only and is not required to be operational for fuel movement. Removal of this portion of the application software will disable the bridge slow zone at the end of bridge travel and the interlock that prevents diagonal motion while in the transfer canal.

The SFPBC rack and cell location digital display and associated slow zone/transfer canal interlocks, are not credited with preventing a Fuel Handling Accident from occurring (UFSAR Section 15.7.4). Removing that portion of the application software does not involve any change to SSCs that adversely affect how UFSAR described design functions are performed or controlled. The proposed activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR because it is not an initiator of any accident, and it creates no new accident initiators. As such, the result and consequences of a malfunction or failure are not different than previously analyzed in the UFSAR and are therefore bounded by existing UFSAR analyses. Therefore, the proposed activity does not change the frequency of occurrence of any accidents previously evaluated in the UFSAR.

This activity is expected to resolve chronic problems with microprocessor failures. The position indication function of the software code requires periodically writing position information into a file, which is stored in the microprocessor memory location. Errors in writing position information into a file, which is stored in the microprocessor memory location, have caused corruption of the executable program. Removal of the position indexing software code should eliminate the file writing activity and eliminate the periodic corruption of the associated tracking file and application program. Therefore, the premise of this activity is to decrease the likelihood of equipment malfunction.

The non-safety related Spent Fuel Pool Bridge Crane rack and cell location digital display are not credited in a Fuel Handling Accident (UFSAR Section 15.7.4). The physical events accompanying a fuel handling accident do not prevent correct functioning of the controls and instrumentation of other safety systems, which provide active safety features following an accident, such as the Emergency Core Cooling System, emergency power, containment integrity, etc. Therefore, this activity will not increase the consequences of an accident previously evaluated in the UFSAR.

Removing that portion of the application software does not involve any change to SSCs that adversely affect how UFSAR described design functions are performed or controlled. This activity should resolve chronic problems with microprocessor failures. Therefore, the consequences of malfunction of an SSC important to safety previously evaluated in the UFSAR will not be increased.

This activity will install revised software, which will remove the position information displayed on the operator's pendant. Removal of this portion of the application software will disable the bridge slow zone at the end of bridge travel and the interlock that prevents diagonal motion while in the transfer canal. Administrative procedure controls are in place and require the crane operator to approach travel limits at slow speeds and also travel in one direction only when in the transfer canal. The administrative controls, which apply both with the indexing system installed and also with the indexing system removed, will remain in place. These administrative controls do not rely on the slow zone or canal zone interlocks. The proposed activity does not create the possibility for an accident of a different type than previously evaluated in the UFSAR because it is not an initiator of any accident, and it creates no new accident initiators. Therefore, the proposed activity will not create possibility for an accident of a different type than previously evaluated in the UFSR.

50.59 REVIEW COVERSHEET FORM

Station: Byron Unit 0

Activity/Document Number: EC 349308 **Revision Number:** 0

Title: Upgrade to Spent Fuel Pool Bridge Crane 0FH05G Control System; remove position indexing software to free computer memory.

The SFPBC does not physically interact with other SSCs important to safety, in which their failure could cause additional adverse conditions to other equipment. Any malfunction will end in the same result as a malfunction of the existing instrumentation.

Design basis limits for fission product barriers remain unchanged.

UFSAR Section 9.1.4.2.2 currently states that the SFPBC rack and cell location digital display on the pendants is for reference only and is not required to be operational for fuel movement. This activity will install a revision to software in the microprocessor-based Control System, located in the Control Cabinet of the Spent Fuel Pool Bridge Crane (SFPBC). The revised software will remove the position information displayed on the operator's pendant. The SFPBC Rack and Cell location digital display on the pendants is for reference only and is not required to be operational for fuel movement. The SFPBC is operated manually and controlled administratively. There is no change to any of the SFPBC functions. The Operator's control of the SFPBC and emergency shutdown of the SFPBC remain the same. For this reason, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the UFSAR.

This activity does not impact operation of any systems specified in the Technical Specifications. No Technical Specification or Operating License change is required for this activity. However, an UFSAR change is required to revise section 9.1.4.2.2 and 9.1.4.3.1. DRP 11-002 has been initiated for this change. This change will reflect removal of the position information displayed on the operator's pendant and the interlock that prevents diagonal motion while in the transfer canal.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
X	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-04-0008**	**Rev.**	**0**

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): BYRON UNIT 2

Activity/Document Number: EC 354059 **Revision Number:** 0

Title: Zinc Injection Into The CV Letdown Line and Reactor Coolant Systems

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

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

The proposed activity installs a non-safety related vendor supplied zinc acetate chemical feed skid consisting of pumps, tanks, valves, and controls in the Auxiliary Building Unit 1 Containment Chiller Room. Along with the skid, necessary tubing, tubetrack, hangers, valves and power (120VAC supplied by a local lighting panel) will be installed. Additionally, some existing abandoned tubing will be removed and existing hangers will be modified.

The new 3/8" tubing will connect to the Chemical and Volume Control System (CV) System on the letdown line. The skid will feed the zinc solution through the CV System into the Unit 2 Reactor Coolant System (RCS). The safety classification of Line 2CVG2B-3/8 (ASME Class 3) will be changed to non-safety and Valve 2CV8129 will be changed from normally open to normally closed.

Upon completion of the installation, the soluble zinc acetate compound will be injected into the RCS during the later part of Byron Unit 2 Cycle 12. The goal is to establish a reactor coolant zinc concentration of 10 ppb during Cycle 12. A higher concentration during future cycles is limited to 40 ppb. All evaluations for zinc injection have been performed for 40 ppb for an eighteen-month fuel cycle, with temporary excursions of 80 ppb found acceptable. The analyses performed in support of this activity will allow the use of zinc injection for all future cycles with respect to fuel and non-fuel issues. With each fuel reload, cycle specific fuel rod design analyses are performed and are documented in the cycle specific Reload Safety Evaluation Report. The impact of zinc addition will be addressed in the cycle specific Reload Safety Evaluation Report for each future cycle. These cycle specific analyses will ensure that all fuel rod design criteria are met with due consideration of zinc addition effects.

The UFSAR will be updated (DRP 11-026) to discuss zinc injection into the Reactor Coolant System. The Fire Protection Report will be updated (FDRP 22-012) to reflect the installation of the zinc skid in the Unit 1 Auxiliary Building on El. 401'.

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

Zinc injection is employed at numerous PWR plants similar to Byron Station. At low concentrations the zinc replaces radioactive cobalt, iron, and nickel in RCS component corrosion oxide layers. Over time, this reduces radiation fields and subsequently personnel radiation dose. At higher concentrations, the zinc is known to inhibit general corrosion and initiation of primary water stress corrosion cracking (PWSCC) of Alloy 600 materials (e.g., Steam Generator tubes). Extensive testing, analysis, and operating experience at other plants has demonstrated the benefits of zinc injection without detrimental effects. Therefore, the decision has been made to implement zinc injection at Byron Station.

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

With the exception of the following, all other technical issues involved with this activity have been addressed as having no impact on plant operations, design basis, or safety analysis as described in the UFSAR.

For RCS non-fuel components, the injection of zinc will cause the following effects: The ionic zinc will replace cobalt, nickel, and iron in the oxidation layers on components in the RCS. In the short term, these radionuclides will enter the coolant, causing radiation levels within the Containment and Auxiliary Building (primarily CV letdown piping and the Mixed Bed Demineralizers) to rise. In the long term, after a few fuel cycles and based on other plant experiences, the radiation levels within the plant will be reduced to below pre-zinc injection levels. The second

Station/Unit(s): BYRON UNIT 2

Activity/Document Number: EC 354059 **Revision Number:** 0

Title: Zinc Injection Into The CV Letdown Line and Reactor Coolant Systems

effect of the zinc in higher concentrations will be to inhibit initiation of PWSCC. That is, testing has demonstrated that zinc is effective in reducing PWSCC initiation in Alloy 600 and other austenitic steels. The Steam Generator tubes at Byron 2 are composed of Alloy 600 material. Additionally, zinc has been demonstrated to be effective in reducing PWSCC corrosion rates of wetted RCS surfaces. Therefore, the use of zinc may be beneficial in reducing the possibility of corrosion-induced RCS pressure boundary failures. No detrimental effects on equipment or components (such as pump seals, heat exchanger performance) have been reported at plants using zinc injection.

During refueling outages, reactor coolant water will mix with refueling water. That is, when the reactor coolant water is mixed with water from the Refueling Water Storage Tanks (RWST) and the Spent Fuel Pool in the reactor cavity, some zinc will be transported into the Residual Heat Removal (RH) System, the Spent Fuel Pool, and the RWST. Because zinc injection will be terminated prior to a normal Unit shutdown and the large differences in volume between the RCS and RWST, the amount of zinc in the non-RCS systems will be insignificant. Additionally, zinc will migrate from spent fuel into the Spent Fuel Pool water. Since the equipment, components and materials in these systems are similar to those in the RCS, the zinc will have no adverse affect on these systems or their operation.

The CV Mixed Bed Demineralizers will remove the liberated radionuclides in the coolant. This likely will result in higher demineralizer dose levels and possibly a reduction in mixed bed ion exchange resin longevity. However, all zinc addition plants discovered that they could operate for a complete fuel cycle on a single resin bed charge. Likewise, filters (such as the Reactor Coolant Filter) may experience higher levels of contamination and/or require more frequent replacement. The plant demineralizers and filters will be monitored using existing plant equipment and procedures and preparations put in place to readily change-out resins and/or filters if the need arises. Based on industry experience, changes in radiation levels and demineralizer/filter differential pressure has typically been gradual.

As stated above, zinc injection will be terminated prior to a *planned* Unit shutdown or subsequent to an *unplanned* shutdown. This is because of the retrograde solubility of zinc with temperature. As the RCS temperature is reduced, zinc concentration levels in the coolant will tend to increase. A zinc concentration of 80 ppb (twice the maximum allowable concentration for Mode 1 operation) for short excursions has been analyzed and found acceptable. Increases above 80 ppb for Modes other than Mode 1 are acceptable.

Industry operating experience indicates there will be an insignificant effect on Crud Induced Power Shift (CIPS) due to zinc injection on the corrosion of ZIRLCO fuel cladding in Units that have low and medium duty cores. While the magnitude of CIPS due to zinc addition induced crud deposits on the core is unknown for a high duty core (Byron), an evaluation was performed to predict oxide layer (crud) thickness. The evaluation conservatively bounds any increases in corrosion thickness that can be expected due to zinc addition.

To mitigate risk associated with CIPS, nickel concentrations and axial power distribution will be monitored throughout the cycle. Chemistry procedures have been developed to monitor RCS chemistry for zinc addition. Existing flux mapping surveillances will be employed to measure axial power distribution. Excessive coolant nickel levels or indications of CIPS may result in the cessation of zinc addition.

Based on the above limitations and monitoring, zinc injection will not cause fuel cladding corrosion levels to exceed their design parameters.

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

The proposed zinc injection into the Reactor Coolant System will not result in a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR since the zinc has no adverse affect on pressure

Station/Unit(s): BYRON UNIT 2

Activity/Document Number: EC 354059 **Revision Number:** 0

Title: Zinc Injection Into The CV Letdown Line and Reactor Coolant Systems

boundary integrity, the small flow and volume of zinc solution does not increase operator response time during boron dilution events, the amount of concentrated zinc solution used for injection is small and has no affect in case of a spill on Main Control Room habitability, the new installation is designed and installed to the appropriate codes and standards to prevent any interaction with safety related equipment and the possibility of increased corrosion on the fuel is limited by the chemistry sampling and core monitoring to prevent oxidation from reaching design limits. Therefore, there is no increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The proposed zinc injection will not result in a minimal increase in the likelihood of occurrence of a malfunction of an SSC import to safety previously evaluated in the UFSAR since there is no or insignificant affect on hydrogen generation, ECCS water pH and Recirculation Sump blockage; there are no adverse affects on any equipment or components in the RCS or ECCS or their performance; fuel oxidation thicknesses remain within design parameters; and procedures developed to control zinc injection do not interface with plant operation.

The proposed zinc injection will not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR since there is no adverse affect on any fission product barrier, the release associated with the zinc oxide layer during a postulated accident would be a small fraction of the total release, zinc injection does not change the response of the RCS and engineered safeguard systems to postulated accident conditions, there is no increase in post-LOCA hydrogen generation in the Containment, there is no increase in the potential for Recirculation Sump blockage, there is no change in sump water pH, there is no increase in operator response time to a boron dilution event, during normal operation a break in the new tubing is bounded by existing analysis, during a LOCA, the new tubing is isolated on a Containment isolation signal, there is no affect on any Chapter 15 accidents. Therefore, zinc injection does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

Zinc injection will not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR since there is no equipment malfunction that would result in a larger release to the environment, there is no affect on systems and components used to mitigate the consequences of postulated accidents, no fission product barriers are degraded, radiation release both planned and during accidents have been evaluated any increase due to zinc injection is bounded by existing analyses, and no affect on equipment qualification due to increase radiation levels in the plant. Therefore, zinc injection does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Zinc addition will not create a possibility for an accident of a different type than any previously evaluated in the UFSAR since zinc is not the initiator of any accident, zinc does not introduce a new failure mode in fission product barriers, zinc will not change the manner in which the plant is operated, existing pipe break analyses bound any postulated breaks, zinc does no alter the response of the RCS or engineered safeguards system to transient conditions and there is no affect on Main Control Room habitability.

Zinc injection will not create a possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the UFSAR since there are no postulated malfunctions, there are no adverse impacts on the operation or function of any SSCs and all new components have been designed to meet existing plant criteria.

Zinc addition will not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered since the oxide layer altered by zinc is thinner and inhibits stress corrosion cracks and the fuel will be monitored such that the design basis parameters are not exceeded.

The methods of evaluation described in the UFSAR used in establishing the design basis and safety analysis were employed to support the conclusions drawn in evaluating the use of zinc. Additionally, the qualification of the components used in the installation was performed consistent with the UFSAR.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): BYRON UNIT 2

Activity/Document Number: EC 354059 **Revision Number:** 0

Title: Zinc Injection Into The CV Letdown Line and Reactor Coolant Systems

With the presently performed fuel and non-fuel analyses and with the limitations, monitoring and/or testing imposed by future cycle specific Reload Safety Evaluation Reports, zinc injection can be implemented without further evaluation.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
X	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-05-0001**	**Rev.**	**0**

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number:
EC 356569 (BY), EC 356733 (BR), DRP 11-086, and Affected Procedures (See below) **Revision Number:** 1,1,0

Title: **Replacement of the Containment Recirculation Sump Screens**

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

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

Background
The containment recirculation sumps (two) are located at elevation 377 feet in the containment, the lowest floor elevation in the containment exclusive of the reactor cavity. The containment recirculation sumps are the source of water for the Emergency Core Cooling System (ECCS) and the Containment Spray (CS) System during Recirculation Phase following a Loss Of Coolant Accident (LOCA). In response to a LOCA, the residual heat removal (RHR), centrifugal charging (CV) pumps, and Safety Injection (SI) System pumps automatically start upon receipt of a safety injection signal and inject to the reactor coolant system, taking suction from the refueling water storage tank (RWST). The containment spray pumps start automatically when the containment pressure set point is reached and also take suction from the RWST. The switchover to the containment recirculation sumps as suction source to the RHR pumps is initiated when the RWST water level decreases to 46.7%. The CS pumps suction source is switched over to the containment recirculation sumps when the RWST water level decreases further to 12%.

The existing design for the containment recirculation sumps includes a set of three screens to protect the intake pipe for each sump. An inner screen is located within the sump with the two outer screens located above elevation 377 feet.

The U.S. Nuclear Regulatory Commission (NRC) has issued Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004 to request that addressees perform an evaluation of the ECCS and CS recirculation functions in light of the information provided in the GL and, if appropriate, take additional actions to ensure system function. The request was based on identified potential susceptibility of the pressurized water reactor (PWR) recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of ECCS or CS and on the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage. The nuclear industry has developed a methodology (Document #NEI 04-07) to perform the required evaluations with the cooperation of the Nuclear Energy Institute (NEI) and the NRC has issued a Safety Evaluation Report on the NEI document in December of 2005. Modifications required to the existing screens must be installed by December 31, 2007.

The most important elements of the NEI methodology are break selection, debris generation, and debris transport. Substantial conservatism is built into the evaluation process. For example, although application of the Leak-before-Break methodology has been approved for Byron and Braidwood on the main Reactor Coolant Loop piping, the debris generation analysis assumes the largest pipes in the Reactor Coolant System break. Byron and Braidwood have followed the NEI methodology. The results of the evaluation concluded that the existing screens must be replaced. Control Components Inc. (CCI) in Switzerland has designed and manufactured the replacement screen assemblies.

In addition, chemical reactions between the post-LOCA water and materials (Aluminum, copper, concrete) located in the containment building may result in chemical precipitants forming when the post-LOCA water temperature decreases in the latter part of the accident. This is called "chemical effects"; the design of the replacement screens is required to incorporate the impact on head loss due to "chemical effects".

Head loss testing on a scaled version of the replacement screens has been performed using scaled quantities of debris (coating, fiber insulation, reflective metal insulation, glass). The replacement screen assemblies that will be installed per EC 356569 and EC 356733 have been verified to have a head loss that maintains adequate margin for the Net Positive Suction Head (NPSH) for the RHR and CS pumps when they take suction from the containment recirculation sumps. The head loss testing performed by CCI also includes chemical effects.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number:
EC 356569 (BY), EC 356733 (BR), DRP 11-086, and Affected Procedures (See below) **Revision Number:** 1,1,0

Title: **Replacement of the Containment Recirculation Sump Screens**

Modification

This activity will remove the existing screens (within the sump pits and at elevation 377 ft) for the Containment Recirculation Sumps A and B. EC 356569 and EC 356733 will install a new screen assembly that has been designed in accordance with the NRC methodology that is discussed in Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," and related correspondence. The filtering sections of the new screen assemblies are perforated, stainless steel plates with circular openings of 1/12 inch in diameter. The approximate surface area of the replacement screens is 3,000 ft^2 per sump. The outer screen from the current design of the containment recirculation sumps will be replaced with stainless steel grating (with nominal openings size of 1"x2") which will make up the trash rack. The function of the trash rack is to prevent large debris from reaching the recirculation sump intake. A stainless steel plate, maximum height of 2 inches, will be installed at the base of the grating sections; this is needed to prevent operational leakage from entering the containment recirculation sumps where it cannot be quantified.

This activity also removes the bell at the end of the recirculation sump intake pipe and installs a stainless steel ring collar plate that is used for the interface with the water intake assembly located between the strainer banks inside the sump pits.

These ECs also include the removal of the containment recirculation sump pit level indicating switches 1LS-0940A and 1LS-0941A for Byron Unit 1, 2LS-0940A and 2LS-0941A for Braidwood Unit 2, along with the associated level indicating lights 1LL-SI075A and 1LL-SI075B for Byron Unit 1, 2LL-SI075A and 2LL-SI075B for Braidwood Unit 2 from the main control room.

UFSAR Changes

UFSAR Change Package 11-086 revises the UFSAR to reflect the installation of the replacement screen assembly and to reflect the removal of the ECCS Sump Level Instrumentation.

Procedure Changes (Byron)

- 1BEP-0, Revision 108, Page 24, Step 29, 3rd Bullet
  Remove "Cnmt sump level lights" and replace with "RF sump level indication".
- 1BOA PRI-7, Revision 104, Page 15, Step 5.d, 1st Bullet
  Remove "Cnmt sump level lights"
- 1BOA PRI-1, Revision 103, Page 1, Step B.1
  Delete the 9th Bullet (CNMT recirc sump level rising)
- 1BCA 1.3, Revision 001
  Remove the entry conditions that reference the sump level lights.
- BOP AP-73A1, Revision 004
  Remove containment sump level lights from list of instruments that lose power.
- BOP AP-74A1, Revision 003
  Remove containment sump level lights from list of instruments that lose power.

Procedure Changes (Braidwood)

- 2BwCA 1.3, Revision 001
  Remove the entry conditions that reference the sump level lights.
- 2BwOA ELEC-2, Revision 101
  Remove the Sump Level lights from the list of instruments that lose power.
- 2BwOA PRI-1, Revision 103, Page 1, Step B.1
  Delete the 9th Bullet (CNMT recirc sump level rising) from the list of entry conditions.
- BwOP IP-2T1, Revision 004
  Remove containment recirculation sump light box (2LL-SI075A) from list of instruments that will not illuminate.
- BwOP IP-2T4, Revision 004
  Remove containment recirculation sump light box (2LL-SI075B) from list of instruments that will not illuminate.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number:
EC 356569 (BY), EC 356733 (BR), DRP 11-086, and Affected Procedures (See below) **Revision Number:** 1,1,0

Title: **Replacement of the Containment Recirculation Sump Screens**

The Containment Recirculation Sump lights are referenced in the Severe Accident Guidelines Procedures (SAMG) as indication of containment water level and indication of ECCS in recirculation mode. These references will be removed to reflect the changes made via EC 356569 and 356733. The Severe Accident Guidelines Procedures also list the containment floor level instruments (_LI-PC006 and _LI-PC007) for indication of flooding levels inside containment. These instruments will remain available. These procedures are beyond design basis and will not be addressed in this 50.59 review.

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

This activity installs a new passive screen design on Byron Unit 1 and Braidwood Unit 2 to support the operation of the pumps taking suction from the containment recirculation sumps. The acceptable operation of the new screens has been verified with the screens loaded with post-accident design basis debris. This modification does not resolve all issues associated with Generic Letter 2004-02. Outstanding issues related to resolution of blockage of downstream components will be resolved by implementing other modifications that will be addressed during subsequent activities.

The physical configuration of the replacement strainer in the sump pits interferes with the existing level instrumentation assembly. This level instrumentation and the related indicating lights in the Main Control Room are removed via these ECs.

Discussion of 50.59 Reviews
These ECs include the removal of the containment recirculation sump pit level indicating switches 1LS-0940A and 1LS-0941A for Byron Unit 1, 2LS-0940A and 2LS-0941A for Braidwood Unit 2, along with the associated level indicating lights 1LL-SI075A and 1LL-SI075B for Byron Unit 1, 2LL-SI075A and 2LL-SI075B for Braidwood Unit 2 from the main control room. This action has an adverse impact on the design and function of these switches as described in UFSAR Sections 6.2.1.7, 6.3.1, 6.3.5.4 & 7.7.1.20. Removal of the instruments results in the deletion of a function described in the UFSAR, therefore, a 50.59 Evaluation will be prepared for the removal of these level switches and associated indicating lights.

The methodology for evaluating the adequacy of the NPSH to the impacted pumps is described in UFSAR Section 6.3.2.2. The NPSH Available is calculated using a saturated sump model in which, the containment atmospheric pressure is conservatively assumed to be equal to the vapor pressure of the liquid in the containment recirculation sumps. This assumption ensures that credit is not taken for containment pressurization during the transient. The same saturated sump model is also used for the NPSH analysis at design temperature (200°F). As part of the tasks required to disposition the chemical effects on head loss through the strainers, the NPSH analysis has been performed at low temperatures. As demonstrated by the NRC and Industry testing, the precipitants due to the chemical effects do form as the recirculation sump fluid cools down. The NPSH analysis at low temperatures does not use the saturated sump model. This is a deviation from the methodology described in the UFSAR. This change in methodology will be addressed in question 8 of the 50.59 Evaluation.

Summary:

- 50.59 Screening –
  Addresses the modification to replace the containment recirculation sump screens along with all changes to the UFSAR resulting from the modification.

- 50.59 Evaluation –
  Addresses the removal of the containment recirculation sump level indicating switches and indicating lights in the main control room. The response to Question 8 also addresses the change in methodology from the NPSH analysis.

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

This activity does not impact plant operations. The replacement screens for the containment recirculation sumps perform the same function as the existing screens. The flow-path of the filtered post-accident water will continue to be to the RHR and CS pumps. The only difference is that the replacement screens have been designed to have acceptable head loss under design basis

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

Activity/Document Number:
EC 356569 (BY), EC 356733 (BR), DRP 11-086, and Affected Procedures (See below) **Revision Number:** 1,1,0

Title: **Replacement of the Containment Recirculation Sump Screens**

debris loading conditions. The new screens within the sumps as well as the modified trash rack screens are passive devices and require no changes to existing system operation.

The performance of the replacement screens has been verified by analysis and testing. The replacement screens have been tested by the OEM under design basis debris loading conditions. The impact of the resulting head loss on the Net Positive Suction Head (NPSH) Available for the CS and RHR pumps has been evaluated. The NPSH for the CS and RHR pumps remains acceptable. The ECCS and CS System are thus capable of performing their design basis functions as described in the UFSAR. The safety analyses described in the UFSAR that assume the operation of the RHR and CS systems in the recirculation mode are thus not impacted by this modification.

The elimination of the level switches (_LS-0940A and _LS-0941A) and level lights (LL-SI075A and _LL-SI075B) removes one method to diagnose the progression of a LOCA. However, the instruments are not environmentally qualified and, therefore, while they are included in the Emergency Operating Procedures (EOPs), they are not credited in the UFSAR as the means to verify adequate water volume exists in containment prior to performing the complete sequence to transfer the suction of the pumps to the containment recirculation sumps.

The changes to Byron Station procedures 1BEP_0 and 1BOA PRI-7 remove references to the ECCS sump level lights. The RF sump level indication is added in place of the ECCS sump lights. The replacement instrumentation provides adequate data that can be used to diagnose if the RCS is intact. The change to 1BOA PRI-1 removes a reference to an entry condition that is no longer applicable.

The lights associated with the ECCS sump level instrumentation are credited for indicating the potential for spurious operation of valve 2SI8811A&B during a fire in certain (11.3-2) fire zones. The Fire Protection Safe Shutdown Report and associated procedures will be modified to remove references to the instruments being removed. Changes to the Fire Protection Program are addressed as part of a separate documentation change package for the Fire Protection Report (FDRP 22-047 for Braidwood and FDRP 22-049 for Byron).

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

Screening Summary
The installation of the replacement screen assembly does not degrade the performance of the RHR and CS pumps that take suction from the containment recirculation sump. The ECCS and CS System are thus capable of performing their design basis functions as described in the UFSAR.

None of the procedure changes have an adverse effect on how plant design functions described in the UFSAR are performed or controlled. The methodology that was used to develop the design for the replacement strainer has been endorsed by the NRC and does not change an element of a methodology that is described in the UFSAR. This activity does not involve a test or experiment. No changes are necessary to the Technical Specifications or the Operating License.

50.59 Evaluation Summary
Removal of the containment recirculation sump level instruments and indicating lights does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR. Removal of these lights does not result in a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. Removal of the instruments does not increase the consequences of any accident since all barriers credited in limiting the consequences of an accident are maintained within their acceptance criteria as stated in the UFSAR. Since the procedure changes do not change the response sequence to a design basis accident, the changes do not increase the consequences of the accident. Removal of the containment recirculation sump level instruments does not result in malfunctions of other components nor does it create the possibility of an accident of a different type than previously evaluated in the UFSAR.

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number:
EC 356569 (BY), EC 356733 (BR), DRP 11-086, and Affected Procedures (See below) **Revision Number:** 1,1,0

Title: Replacement of the Containment Recirculation Sump Screens

This activity will not degrade the function of the ECCS systems for long term core cooling post-LOCA since the replacement screens have been evaluated to maintain adequate NPSH for the RHR and CS pumps when they take suction from the containment recirculation sump.

The UFSAR does not provide a description of the methodology used in the design of the current screens. The design of the new screens was done in accordance with NRC approved methodology for resolution of this generic safety issue. NPSH calculations were performed in accordance with the UFSAR described methodology and the new NRC requirements to confirm that the pumps will have adequate NPSH available.

The removal of the containment recirculation sump level instrumentation and corresponding indicating lights was considered adverse to the function of these components as contained in the UFSAR Sections 6.3.1 & 6.3.5.4 and a 50.59 Evaluation was performed to address the removal of these components. The evaluation determined that the removal of the instrument does not increase the probability of an accident or the consequences of an accident. The removal of the components has no adverse impact on the function, performance or qualification of any other plant component and cannot cause the malfunction of any SSC Important to Safety.

The NPSH evaluation at low temperatures required using a methodology that is different than the saturation model described in the UFSAR. The methodology that was used is in accordance with the NRC guidelines given in Safety Guide 1 for evaluating NPSH for the RHR and CS pumps; therefore NRC approval is not necessary.

Results of the 50.59 Review
This activity can proceed based on the governing procedure without prior NRC approval.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

	Applicability Review				
X	**50.59 Screening**	**50.59 Screening No.**	6E-06-0108 (BY EC) BRW-S-2006-188 (BR)	**Rev.**	0 0
X	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-06-0003 (BY) BRW-E-2006-187 (BR)	**Rev.**	0 0

Procedure Revisions and Special Process Procedures (SPP)	
1.	6G-04-009 Rev. 0
2.	6G-05-0002 Rev. 0

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 1

Activity/Document Number: SPP 04-012 Power Ascension Testing **Revision Number:** 0

Title: Byron Unit 1 Power Ascension DEHC Testing - SPP 04-012 (Section 6) & Procedure 1BGP 100-3

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

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

EC # 343598 replaces the existing Westinghouse Mod II Digital Electro-Hydraulic (DEH) control system for the main turbine with a redundant fault tolerant Westinghouse Ovation based Distributed Control System (DCS).

SPP 04-012 is used to verify the functionality of the new digital control system. SPP 04-012 Section 1 is performed during site acceptance testing prior to installation in the plant. SPP 04-012 Sections 2 through 5 performs modification testing during Modes 5 or 6 prior to plant startup.

This activity (Byron Unit 1 Power Ascension DEHC Testing) utilizes SPP 04-012 Section 6 and will perform the required modification testing during Byron Unit 1 Cycle 14 initial power ascension, from Startup (Mode 2) to Power Operation (Mode 1), of the new digital control system. This testing will be performed utilizing sections added to procedure 1BGP 100-3 to support the required testing during initial power ascension as delineated on modified 100-3T1 flow chart, as follows:

- Following 1BGP 100-3 Step F.5 (Placing the Reheat Temperature Controller in AUTO) and prior to 1BGP 100-3 Step F.7 (Main Turbine Speed Increase to 520 RPM);
  - Test Emergency Trip system (ETS) Speed Circuitry; Tripping Turbine at 0 rpm
  - Test Operator Auto/Overspeed Protection Controller (OA/OPC) Overspeed Circuitry; Tripping Turbine at 0 rpm
- Following 1BGP 100-3 Step F.14 (Turbine Speed Increase to 1700 RPM and Transfer to GV Control) and prior to 1BGP 100-3 Step F.15 (Turbine Speed Increase to 1800 RPM);
  - Transfer to Manual and Restore Back to Auto Control In Speed Control
- Following 1BGP 100-3 Step F.33 (Begin Generator Loading to 125 MW);
  - Increase to 250 MW (20%) with Megawatt Control Loop IN Service
  - Transfer to Manual Mode Operation and Back To AUTO While In Load Control
- Following 1BGP 100-3 Step F.42 (Placing the FW Pump Controllers in Automatic if required) and prior to 1BGP 100-3 Step F.44 (Transfer from Single Valve to Sequential Valve Mode of Control for Turbine-Generator Governor Valves);
  - Transfer to Sequential Valve
  - Load Step Change While in Single Valve Control
- Following 1BGP 100-3 Step F.44 (Transfer from Single Valve to Sequential Valve Mode of Control for Turbine-Generator Governor Valves);
  - Load Step change While In Sequential Valve Control
- Following 1BGP 100-3 Step F.63 (Increase Turbine Load to 1120 MW, 90%);
  - Governor Valve Test and Tuning
  - Throttle Valve test
  - Restore Sequential Valve Mode

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

Safety Evaluation 6G-04-0003 evaluated the modification (EC # 343598) to replace the DEH Turbine Controls system. In normal circumstances the modification 50.59 is used and no additional 50.59 is performed, which is allowed by 50.59 resource manual per 10CFR50.65(a)(4). However, in this case, part of the modification test is being done during power ascension test to dynamically test the system. This requires separate 50.59 because the test is being done with operating plant.

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

This activity will perform power ascension testing of the new digital control system during Byron Unit 1 Cycle 14 initial power ascension, from Startup (Mode 2) to Power Operation (Mode 1). This test verifies the system controls the turbine from roll-up to full power Speed and Load Control functions, Valve Test timing, Valve loop controls are verified to operate correctly and will be tuned to Byron Unit 1 turbine components to optimize plant performance.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 1

Activity/Document Number: SPP 04-012 Power Ascension Testing **Revision Number:** 0

Title: Byron Unit 1 Power Ascension DEHC Testing - SPP 04-012 (Section 6) & Procedure 1BGP 100-3

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

EC # 343598 replaces the DEH Turbine Controls system with a more reliable system that uses logic based on the existing plant design updated to current control system technology.

The functionality of the new digital control system has been verified during factory acceptance testing. Site acceptance testing (SPP 04-012 Section 1) will re-verify functionality of the new digital control system prior to installation in the plant. Modification testing during Mode 5 or 6 (SPP 04-012 Sections 2 through 5) prior to plant startup will verify proper installation and system integration. This activity will perform power ascension testing of the new digital control system utilizing SPP 04-012 Section 6 during Byron Unit 1 Cycle 14 initial power ascension. Step changes during power ascension will be at low power.

Westinghouse has reviewed this testing procedure and indicates that that the testing sequence does not create an adverse condition for the fuel or the core during or after the testing sequences. From e-mail dated 11/22/04 from Jeffrey Guthridge (Westinghouse) to R. Niederer (Exelon), " Power level for DEHC testing is below threshold power [40%], therefore PCMI [Pellet Clad Mechanical Interaction] is minimal. Although power changes proposed are rapid they are small overall, no significant impact on fuel."

This activity will perform dynamic testing of the Throttle valve (TV) / Governor Valve (GV) above 90% power to optimize plant performance. The DEH modification does not impact the actuators or valve characteristics of any valves associated with turbine control. Valve response times (UFSAR Table 10.2-1) and control functions are not impacted by this change.

Power ascension testing of the new digital control system performed by this activity will not result in any system or component required for accident or transient mitigation to be operated, shutdown, or isolated such as to impact its ability to perform its safety related mitigating functions. As such, the result and consequences of a malfunction or failure are not different than previously analyzed in the UFSAR and are therefore bounded by existing UFSAR analyses. Therefore, the proposed activity does not change the frequency of occurrence of any accidents previously evaluated in the UFSAR.

This activity will perform testing of the new digital control system utilizing SPP 04-012 Section 6 during Byron Unit 1 Cycle 14 initial power ascension. A Heightened Level of Awareness (HLA) briefing will be performed with all personnel involved with the power ascension testing of the new digital control system. All plant and equipment operation and surveillance will be performed in accordance with approved plant procedures. Therefore, the premise of this activity is to decrease the likelihood of equipment malfunction. With the properties and attributes of the SSCs important to safety remaining consistent with existing instrumentation design and detection requirements, the activity will not increase in the likelihood of a malfunction of an SSC important to safety as previously evaluated in the UFSAR.

Power ascension testing of the new digital control system performed by this activity will not change the configuration, operation and accident response of any system or component such as to impact its ability to perform its safety related function. The incorporation of the new digital components and logic provides a more reliable system. The modification does not impact the closure time or control function of any turbine valves. The system is non-safety related and response times are not credited in any of the accident analyses. Therefore, this activity will not increase the consequences of an accident previously evaluated in the UFSAR.

The incorporation of the new digital components and logic is a reliability enhancement to an existing system described in the UFSAR. The system is non-safety related and the modification does not change the function or performance requirements for the system as discussed in the UFSAR. The operation of equipment during Byron Unit 1 Cycle 14 initial power ascension will be monitored to ensure the equipment is operating within design basis and operating limits. As such, the result and consequences of a malfunction or failure are not different than previously analyzed in the UFSAR and are therefore bounded by existing UFSAR analyses. Therefore, with the properties and attributes of the SSCs important to safety remaining consistent with existing instrumentation design and detection requirements, the activity will not increase the consequences of malfunction of an SSC important to safety previously evaluated in the UFSAR.

Power ascension testing of the new digital control system performed by this activity will not result in any system or component required for accident or transient mitigation to be operated, shutdown, or isolated such as to impact its ability to perform its safety related mitigating functions. The configuration, operation and accident response of any system or

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 1

Activity/Document Number: SPP 04-012 Power Ascension Testing **Revision Number:** 0

Title: Byron Unit 1 Power Ascension DEHC Testing - SPP 04-012 (Section 6) & Procedure 1BGP 100-3

component such as to impact its ability to perform its safety related function is unchanged. Therefore, the proposed activity will not create possibility for an accident of a different type than previously evaluated in the UFSAR.

The modified DEH system does not change the function or performance requirements of this non-safety related system as described in the UFSAR, such that important to safety components are required to function in a different manner or environment than currently analyzed in the UFSAR. During the testing of the new digital control system performed by this activity, the operation of equipment during power ascension will be monitored to ensure the equipment is operating within design basis and operating limits. Since the function of the new digital control system has been verified prior to installation in the plant during factory acceptance testing / site acceptance testing, and will be verified in modification testing during Mode 5 or 6, the likelihood of a malfunction of the system during Cycle 14 initial power ascension is minimal. The result and consequences of a malfunction is not different than previously analyzed in the UFSAR and are therefore bounded by existing UFSAR analyses. Therefore, the proposed activity will not create the possibility for a malfunction, which ends in a different result than previously evaluated in the UFSAR.

The design basis limits for fission product barriers remain unchanged and the proposed activity is not an initiator of an accident, no new accidents are created. Therefore,

The proposed activity does not result in a departure from a method of evaluation described in the UFSAR that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the UFSAR.

Based on this evaluation, the proposed modification may be implemented without prior NRC approval.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
X	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-04-009	**Rev.**	0

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: BGP 100-3 Power Ascension Testing **Revision Number:** 47

Title: Byron Unit 2 Power Ascension DEHC Testing – Procedure 2BGP 100-3

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

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

EC # 343599 replaces the existing Westinghouse Mod II Digital Electro-Hydraulic (DEH) control system for the main turbine with a redundant fault tolerant Westinghouse Ovation based Distributed Control System (DCS).

SPP 05-003 is used to verify the functionality of the new digital control system. The Site Acceptance Testing is performed prior to installation in the plant. SPP 05-003 Sections 1 through 4 performs modification testing during Modes 5 or 6 prior to plant startup.

This activity (Byron Unit 2 Power Ascension DEHC Testing) utilizes 2BGP 100-3 and will perform the required modification testing during Byron Unit 2 Cycle 13 initial power ascension, from Startup (Mode 2) to Power Operation (Mode 1), of the new digital control system. This testing will be performed utilizing sections added to procedure 2BGP100-3 to support the required testing during initial power ascension as delineated on modified 100-3T1 flow chart, as follows:

- Following 2BGP100-3 Step F.5 (Placing the Reheat Temperature Controller in AUTO) and prior to 2BGP100-3 Step F.7 (Main Turbine Speed Increase to 520 RPM);
  - Test Emergency Trip system (ETS) Speed Circuitry; Tripping Turbine at 0 rpm
  - Test Operator Auto/Overspeed Protection Controller (OA/OPC) Overspeed Circuitry; Tripping Turbine at 0 rpm
- Following 2BGP100-3 Step F.14 (Turbine Speed Increase to 1700 RPM and Transfer to GV Control) and prior to 2BGP100-3 Step F.15 (Turbine Speed Increase to 1775 RPM);
  - Transfer to Manual and Restore Back to Auto Control In Speed Control
- Following 2BGP100-3 Step F.32 (Begin Generator Loading to 120 MW, 10%);
  - Increase to 240 MW (20%) with Megawatt Control Loop IN Service
  - Transfer to Manual Mode Operation and Back To AUTO While In Load Control
- Following 2BGP100-3 Step F.43 (Placing the FW Pump Controllers in Automatic if required) and prior to 2BGP100-3 Step F.43 (Transfer from Single Valve to Sequential Valve Mode of Control for Turbine-Generator Governor Valves);
  - Transfer to Sequential Valve
  - Load Step Change While in Single Valve Control
- Following 2BGP100-3 Step F.43 (Transfer from Single Valve to Sequential Valve Mode of Control for Turbine-Generator Governor Valves);
  - Load Step change While In Sequential Valve Control
- Following 2BGP100-3 Step F.62 (Increase Turbine Load to 1090 MW, 90%);
  - Governor Valve Test and Tuning
  - Throttle Valve test
  - Restore Sequential Valve Mode

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

Safety Evaluation 6G-04-0007 evaluated the modification (EC # 343599) to replace the DEH Turbine Controls system. In normal circumstances the modification 50.59 is used and no additional 50.59 is performed, which is allowed by 50.59 resource manual per 10CFR50.65(a)(4). However, in this case, part of the modification test is being done during power ascension test to dynamically test the system. This requires separate 50.59 because the test is being done with operating plant.

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

This activity will perform power ascension testing of the new digital control system during Byron Unit 2 Cycle 13 initial power ascension, from Startup (Mode 2) to Power Operation (Mode 1). This test verifies the system controls the turbine from roll-up to full power Speed and Load Control functions, Valve Test timing, Valve loop controls are verified to operate correctly and will be tuned to Byron Unit 2 turbine components to optimize plant performance.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: BGP 100-3 Power Ascension Testing **Revision Number:** 47

Title: Byron Unit 2 Power Ascension DEHC Testing – Procedure 2BGP 100-3

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

EC # 343599 replaces the DEH Turbine Controls system with a more reliable system that uses logic based on the existing plant design updated to current control system technology.

The functionality of the new digital control system has been verified during factory acceptance testing. Site acceptance testing will re-verify functionality of the new digital control system prior to installation in the plant. Modification testing during Mode 5 or 6 (SPP 05-003 Sections 1 through 4) prior to plant startup will verify proper installation and system integration. This activity will perform power ascension testing of the new digital control system utilizing 2BGP 100-3 during Byron Unit 2 Cycle 13 initial power ascension. Step changes during power ascension will be at low power.

Westinghouse has reviewed this testing procedure and indicates that that the testing sequence does not create an adverse condition for the fuel or the core during or after the testing sequences. From e-mail dated 11/22/04 from Jeffrey Guthridge (Westinghouse) to R. Niederer (Exelon), " Power level for DEHC testing is below threshold power [40%], therefore PCMI [Pellet Clad Mechanical Interaction] is minimal. Although power changes proposed are rapid they are small overall, no significant impact on fuel."

This activity will perform dynamic testing of the Throttle valve (TV) / Governor Valve (GV) above 90% power to optimize plant performance. The DEH modification does not impact the actuators or valve characteristics of any valves associated with turbine control. Valve response times (UFSAR Table 10.2-1) and control functions are not impacted by this change.

Power ascension testing of the new digital control system performed by this activity will not result in any system or component required for accident or transient mitigation to be operated, shutdown, or isolated such as to impact its ability to perform its safety related mitigating functions. As such, the result and consequences of a malfunction or failure are not different than previously analyzed in the UFSAR and are therefore bounded by existing UFSAR analyses. Therefore, the proposed activity does not change the frequency of occurrence of any accidents previously evaluated in the UFSAR.

This activity will perform testing of the new digital control system utilizing 2BGP 100-3 during Byron Unit 2 Cycle 13 initial power ascension. A Heightened Level of Awareness (HLA) briefing will be performed with all personnel involved with the power ascension testing of the new digital control system. All plant and equipment operation and surveillance will be performed in accordance with approved plant procedures. Therefore, the premise of this activity is to decrease the likelihood of equipment malfunction. With the properties and attributes of the SSCs important to safety remaining consistent with existing instrumentation design and detection requirements, the activity will not increase in the likelihood of a malfunction of an SSC important to safety as previously evaluated in the UFSAR.

Power ascension testing of the new digital control system performed by this activity will not change the configuration, operation and accident response of any system or component such as to impact its ability to perform its safety related function. The incorporation of the new digital components and logic provides a more reliable system. The modification does not impact the closure time or control function of any turbine valves. The system is non-safety related and response times are not credited in any of the accident analyses. Therefore, this activity will not increase the consequences of an accident previously evaluated in the UFSAR.

The incorporation of the new digital components and logic is a reliability enhancement to an existing system described in the UFSAR. The system is non-safety related and the modification does not change the function or performance requirements for the system as discussed in the UFSAR. The operation of equipment during Byron Unit 2 Cycle 13 initial power ascension will be monitored to ensure the equipment is operating within design basis and operating limits. As such, the result and consequences of a malfunction or failure are not different than previously analyzed in the UFSAR and are therefore bounded by existing UFSAR analyses. Therefore, with the properties and attributes of the SSCs important to safety remaining consistent with existing instrumentation design and detection requirements, the activity will not increase the consequences of malfunction of an SSC important to safety previously evaluated in the UFSAR.

Power ascension testing of the new digital control system performed by this activity will not result in any system or component required for accident or transient mitigation to be operated, shutdown, or isolated such as to impact its ability to perform its safety related mitigating functions. The configuration, operation and accident response of any system or component such as to impact its ability to perform its safety related function is unchanged. Therefore, the proposed activity will not create possibility for an accident of a different type than previously evaluated in the UFSAR.

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron Unit 2

Activity/Document Number: BGP 100-3 Power Ascension Testing **Revision Number:** 47

Title: Byron Unit 2 Power Ascension DEHC Testing – Procedure 2BGP 100-3

The modified DEH system does not change the function or performance requirements of this non-safety related system as described in the UFSAR, such that important to safety components are required to function in a different manner or environment than currently analyzed in the UFSAR. During the testing of the new digital control system performed by this activity, the operation of equipment during power ascension will be monitored to ensure the equipment is operating within design basis and operating limits. Since the function of the new digital control system has been verified prior to installation in the plant during factory acceptance testing / site acceptance testing, and will be verified in modification testing during Mode 5 or 6, the likelihood of a malfunction of the system during Byron Unit 2 Cycle 13 initial power ascension is minimal. The result and consequences of a malfunction is not different than previously analyzed in the UFSAR and are therefore bounded by existing UFSAR analyses. Therefore, the proposed activity will not create the possibility for a malfunction, which ends in a different result than previously evaluated in the UFSAR.

The design basis limits for fission product barriers remain unchanged and the proposed activity is not an initiator of an accident, no new accidents are created. Therefore,

The proposed activity does not result in a departure from a method of evaluation described in the UFSAR that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the UFSAR.

Based on this evaluation, the proposed modification may be implemented without prior NRC approval.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
X	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-05-0002	**Rev.**	0

Technical Requirements Manual (TRM) Changes and Draft Revision Packages (DRP)	
1.	6G-06-0001 Rev. 0

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Byron/Unit 2

Activity/Document Number: Technical Requirements Manual (TRM) Change 06-019 **Revision Number:** N/A

Title: TRM Change 06-019, TRM Section 3.9.a, "Decay Time"

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

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

The following Technical Requirements Manual (TRM) changes will be made to reduce the required Incore Decay Time (ICDT) for B1R14 from 100 hours to 79 hours:

Byron TRM Section 3.9.a, "Decay time," TLCO states "The reactor shall be subcritical for ≥ the last 100 hours (≥ 79 hours for B2R12)." This activity will revise this statement to replace "(≥ 79 hours for B2R12)" with "(≥ 79 hours for B1R14)."

Byron TRM Section 3.9.a Condition A states "Reactor subcritical for < 100 hours (< 79 hours for B2R12)." This activity will revise this statement to replace "(< 79 hours for B2R12)" with "(< 79 hours for B1R14)."

Byron TRM Surveillance TSR 3.9.a.1 states "Verify the reactor subcritical ≥ 100 hours by confirming the date and time of subcriticality. (≥ 79 hours for B2R12)". This activity will change "(≥ 79 hours for B2R12)" to "(≥ 79 hours for B1R14)".

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

It is anticipated that during B1R14, work activities will be completed and the required plant configuration will be established to support commencing movement of irradiated fuel from the reactor vessel to the Spent Fuel Pool (SFP) prior to the current requirement of 100 hours after reactor shutdown. The SFP cooling analysis assumes that fuel transfer begins after 100 hours decay time in the reactor core. This evaluation is being performed to determine if the proposed changes can be made under the provisions of 10 CFR 50.59.

The proposed activity does not address the radiological consequences of a Fuel Handling Accident (FHA). The radiological consequences of a FHA were revised under the Power Uprate (PUR) program using an ICDT of 48 hours. The changes applicable to the FHA did not require review under 50.59, as it was reviewed and approved by the NRC in NRC Letter dated May 4, 2001 to Oliver D. Kingsley (Exelon), Subject: Issuance of Amendments; Increase in Reactor Power, Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2.

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

The proposed activity will allow starting B1R14 fuel offloading activities earlier than the current 100 hours. Occupational dose on the refueling machine may increase slightly.

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

The proposed activity may be implemented without prior NRC review and approval based upon the following:

Changing the ICDT from 100 hours to 79 hours does not change the frequency of an accident because the proposed change does not increase the failure rate of refueling equipment or increase the risk of a FHA due to human error. Spent fuel handling tools will not change, nor will the method/procedures used for handling spent fuel assemblies. The

Station/Unit(s): Byron/Unit 2

Activity/Document Number: Technical Requirements Manual (TRM) Change 06-019 **Revision Number:** N/A

Title: TRM Change 06-019, TRM Section 3.9.a, "Decay Time"

total number of fuel assemblies to be transferred remains the same. There is no effect on the failure probabilities of the SFP cooling system.

Minor Revision 000L to calculation BRW-00-0010-M/BYR2000-007 has been completed to evaluate the impact of changing the ICDT. This calculation accounts for margin remaining in the background decay heat load since the SFP is not filled to capacity. The results of the calculation verified the calculated SFP Bulk Water Temperature, Maximum Local Water Temperature, Maximum Cladding Temperature and Maximum Clad Heat Flux (DNB) remain acceptable.

The design basis SFP criticality analysis (for the SFP re-rack project) assumes a bulk pool water temperature of 4°C (39°F). The proposed change would potentially increase the temperature of the water in the SFP, thus adding negative reactivity. The SFP criticality analysis is therefore not adversely affected.

There are no offsite dose consequences impacted by this change. The ICDT associated with radiological concerns (dropped fuel assembly) has been reduced to 48 hours under the PUR program which has been reviewed and approved by the NRC.

Beginning core alteration and fuel transfer operation as early as 79 hours after shutdown is not expected to significantly increase the occupational dose. UFSAR Tables 12.3-1 and 12.3-2 divide areas in the plant into radiation zones. The design dose rate for each zone is selected to ensure that the exposure limit of 10 CFR 20 is not exceeded. Shielding is established based on ALARA to minimize the dose rate for the selected areas. The areas affected by the defueling operation are designated as High Radiation areas (Zone III) with a design dose rate of >100 mrem/hr. Access to these areas is controlled in accordance with station procedures and radiation work permits. Electronic dosimeters are required to continuously monitor the dose rate in the areas in order to limit personnel exposure to below 10 CFR 20 limits. These existing controls are not affected.

The fission product barriers potentially affected by this change are the fuel clad and the reactor containment. This change does not result in a change to the internal containment pressure that would represent a challenge to the containment design basis limit of 50 psig. Byron Minor Revision 000L to calculation BRW-00-0010-M/BYR2000-007 determined reducing the ICDT to ≥ 79 hours would not exceed the currently assumed SFP temperature limits, nor will it increase the maximum calculated local water temperature, the maximum calculated cladding temperature, and the actual maximum clad heat flux (DNB) beyond acceptable values. Therefore, no design basis limit for a fission product barrier is being exceeded or altered.

The changes made by this activity represent changes in input parameters to the design basis analysis. Decay heat input to the SFP was calculated for the earlier ICDT using the method described in NRC Branch Technical Position ASB 9-2. This is the same method used in the existing analysis. Therefore, this activity does not change the method of evaluation described in the UFSAR or in the Safety Evaluation for the PUR project.

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
X	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-06-0001	**Rev.**	0