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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 2003 and 2004 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 Safety Evaluation Summaries or 50.59 Review Coversheets for changes not previously submitted for the 2000, 2001 and 2002 calendar years. Safety Evaluation Summaries or 50.59 Review Coversheets not previously submitted are those associated with design changes 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,

Stephen Kuczynski

Stephen E. Kuczynski
Site Vice President
Byron Nuclear Generating Station

SEK/MR/rh

Attachment 1, Byron Station 10 CFR 50.59 Report

cc: Regional Administrator - Region III
NRC Senior Resident Inspector - Byron Station

IE47

# Attachment 1

# Byron Station 10 CFR 50.59 Report

10 CFR 50.59 Review Coversheets for Calendar Years 2003 and 2004

and

Safety Evaluation Summaries and 10 CFR 50.59 Review Coversheets not previously submitted for Calendar Years 2000, 2001 and 2002

| Design Change Packages (DCP), Drawing Change Requests (DCR), Engineering Changes (EC), and Temporary Modifications (TMOD) | |
|---|---|
| 1. | 6G-00-0091 |
| 2. | 6G-00-0144 |
| 3. | 6H-02-0022 |
| 4. | 6G-03-0006 |
| 5. | 6G-04-0002, Revision 1 |

Safety Evaluation Summary Form

Tracking No. 6G-00-0091
Activity No. DCP 9600222 Revision 1

DESCRIPTION:
Revision 0 of Design Change Package (DCP) 9600222 provided the design to install blank-off flanges in the auxiliary steam supply to the liquid radwaste evaporators and the auxiliary steam tunnel area. This design will permanently isolate the steam supply to the radwaste evaporators and auxiliary steam tunnel and will isolate the liquid inputs to the evaporators. This design change will abandon in place the evaporators. The High Energy Line Break (HELB) instrumentation to detect and isolate these areas is no longer needed because the steam line is isolated in the turbine building prior to entering the auxiliary steam tunnel and radwaste evaporator rooms. Revision 1 of DCP 9600222 is being issued to remove auxiliary steam isolation switches 0TS-AS031C, D, E, and F and 0TS-AS032C, D, E, and F from the HELB circuits and to revise the TRM to delete auxiliary steam isolation switches 0TS-AS031C, D, E, and F and 0TS-AS032C, D, E, and F from TRM Table T3.3.f-1. The temperature switches will be disconnected and abandoned in place.

SAFETY EVALUATION SUMMARY:

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

   The probability of occurrence of an accident or transient is not increased because the steam line to the radwaste evaporator rooms and auxiliary steam tunnel is isolated in the turbine building prior to entering the auxiliary building. Therefore, the probability of a HELB in the radwaste evaporator rooms and auxiliary steam tunnel has been eliminated. The removal of HELB sensors from these areas does not increase the probability of a HELB. Therefore, the proposed change does not increase the probability of occurrence of any accident or affect their consequences previously analyzed.

   The probability of equipment failures or malfunctions will not be increased because the steam line to the radwaste evaporator rooms and auxiliary steam tunnel is isolated in the turbine building prior to entering the auxiliary building. The probability of a HELB in the radwaste evaporator rooms and auxiliary steam tunnel has been eliminated. The removal of the HELB sensors from these areas does not increase the probability of an equipment malfunction or failure because the source of the high temperature has been eliminated. This change does not create new failures that would increase the probability of a failure of any other components important to safety used in mitigating any accident analyzed in the SAR.

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

Safety Evaluation Summary Form

Tracking No. 6G-00-0091
Activity No. DCP 9600222 Revision 1

The removal of HELB sensors for the areas that not going to experience high temperatures will not increase the possibility of an accident or transient of a different type than previously evaluated because the source of high temperature is permanently isolated and the possibility of a circuit malfunction is reduced due to fewer components. Therefore, the proposed activity will not create the possibility of a different type of malfunction of equipment important to safety than previously evaluated.

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

   The proposed modification does not affect the margin of safety as defined in the Bases since the proposed activity does not change any basis upon which Technical Specifications are based.

Safety Evaluation Summary Form

Tracking No. 6G-00-0144
Activity No. DCR 990487

DESCRIPTION:
Drawing Change Request (DCR) 990487 revises piping schematics M-48-16, 19, and 22 to reflect the as-built configuration of the turbine building equipment (TE) and turbine building floor drain (TF) systems.

SAFETY EVALUATION SUMMARY:

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

   There are no accidents or malfunctions of equipment important to safety described in the SAR that have been determined to be affected by the proposed change. Since this proposed change does not affect the design or operation of any SSC described in the SAR, no credible accident will be affected by this change.

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

   The proposed change for the Turbine Floor and Equipment Drain systems will not create the possibility of an accident or transient of a different type because the drain connections will be normally closed. In the event of a failure of a drain connection, a flooding analysis has been performed for the turbine building using the circulating water as a source. The capacity of the TF and TE tanks are 12,000 gallons which is negligible compared to the amount of water available in the circulating water system.

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

   The TE and TF systems are not described in any Bases for Technical Specifications.

# 50.59 REVIEW COVERSHEET FORM



**Station: Byron**

**Activity/Document Number** EC 79536(U1) / EC337037(U2) **Revision Number:** 0

**Title:** Remove Actuation Signal for Pressurizer PORV 1(2)RY455Afrom the Master Pressurizer Pressure Controller and Use Actual Pressurizer Pressure Signal

**Description of Activity:**
The proposed change will remove the actuation signal for pressurizer Power Operated Relief Valve(PORV) 1(2)RY455A from the demand output of the Master Pressurizer Pressure Controller(MPPC), 1(2)PC-0455A and move it to the actual pressurizer pressure as selected from relay card 1(2)PY/-0455V. This signal currently processed at comparator card 1PB-0455E, which would now receive the actual pressurizer pressure signal directly, instead of the corrected demand output signal from the MPPC. The change will also revise the lift setpoint for 1(2)RY455A from 2335psig to 2345 psig(reset remains at 2315 psig).This change will require internal wiring changes within NSSS control cabinet 1(2)PA05J located in the Auxiliary Electric Equipment Room.

**Reason for Activity:**
PORV 1(2)RY455A currently receives actuation signal through comparator card 1(2)PB0455E that receives signal from the demand output of the MPPC 1(2)PC-0455A. The MPPC compares the actual pressurizer pressure signal with the pressurizer pressure setpoint and produces an output demand signal with an integral action(the corrected demand output signal will vary from the actual pressurizer pressure signal at a greater rate). The setpoint of the comparator card is fixed and is currently scaled to open at the PORV at 87.5% of MPPC output demand which corresponds to +100 psig from Normal Operating Pressure(NOP) of 2235 psig. Therefore, as pressurizer pressure varies, the output demand signal of the MPPC varies at higher rate. This could cause the PORV to lift when actual pressurizer pressure is lower than 2335 psig due to integral action of the MPPC. Thus, the use of the actual pressurizer pressure for actuation of the comparator card will provide the proper control of the lifting of the subject PORV based on actual plant operating conditions. The proposed change would help prevent the rapid cycling of the both PORVs during the water solid pressurizer conditions, thus preventing possible damage.

**Effect of Activity:**
The proposed change will provide more accurate control of the subject PORV by providing the actual pressurizer signal as its lift input. This change does not affect the safety function of the PORV or other associated pressurizer pressure control SSC's. The manual operation of this PORV is not changed. The PORV will continue to operate in the automatic mode , in the same manner as PORV 1(2)RY456, using the actual pressurizer pressure signal instead of the integral demand output of the MPPC. The revised lift setpoint of 2345 psig will maintain the original design feature of having each PORV actuate separately and will help prevent the cycling of these valves under water solid RCS conditions. 1(2)RY456 will maintain its 2335 psig lift setpoint as originally designed. The change will not affect the operation of the proportional heaters, backup heaters and the spray valves, which are controlled from the demand output signal of the MPPC. In addition, this change does not affect the operation or the function of the PORV as used in the Low Temperature Overpressure Protection(LTOP) system.

**Summary of Conclusion for the Activities 50.59 Review:**
50.59 evaluation has concluded (1) the proposed activity does not result in increase in the frequency of occurrence or consequences of an accident previously evaluated in the UFSAR (2) the proposed activity will not increase the malfunction or the consequences of the malfunction of the SSCs important to safety previously evaluated in the UFSAR (3) the proposed activity does not create the possibility of an accident of a different type than previously evaluated in the UFSAR nor it creates the possibility for a malfunction of an SSC important to safety previosly evaluated in the UFSAR (4) The margin of safety as defined in the Technical Specifications is not reduced and therefore the design basis for the fission product barrier as described in the UFSAR is not exeeded and (5) The propsoed activity does not result in a departure from a method of evaluation described in the UFSAR. NFM has evaluated the Pressurizer PORV 1(2)RY-455A Lift Setpoint Change from 2235 psig to 2245 psig per Letter NFM-MW:01-0245 dated 09/10/01, which concluded that the proposed change will have no adverse impact on UFSAR chapter 15 transient analyses. Therefore, NRC notification is not required prior to implementing this change. The change may be implemented in accordance with station approved procedures.

**Attachments:**
Attach completed Applicability Review if 50.59 Screening is not required.
Attach completed 50.59 Screening if 50.59 Evaluation is not required.
Attach completed 50.59 Evaluation if required to be performed.
Attach completed 50.59 Screening and 50.59 Evaluation if multiple discrete elements of an activity have been linked together and certain elements required a 50.59 Evaluation while other elements did not.

**Forms Attached:** (Check all that apply.)

| | | | | | |
|---|---|---|---|---|---|
| ☒ | Applicability Review | | | | |
| ☐ | 50.59 Screening | 50.59 Screening No. | | Rev. | |
| ☐ | 50.59 Evaluation | 50.59 Evaluation No. | | Rev. | |
| ☒ | 50.59 Validation | 50.59 Validation No. | 6H-02-0022 | Rev. | 0 |

# 50.59 REVIEW COVERSHEET FORM



**Station:** Byron

**Activity/Document Number:** EC 341212 **Revision Number:** 0

**Title:** Installation of Temporary Pumps for Draining the 'B' 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 "B" Essential Service Water (SX) System suction piping. The temporary pumps will take suction from the 1B and 2B 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 1B and 2B pumps needs to be drained to facilitate replacement of the suction isolation valves 1/2SX001B. Replacement of the valves requires entering a 72 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 48,000 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 (0SX138B) 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

Station: Byron

Activity/Document Number: EC 341212 Revision Number: 0

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

forms part of the SX system pressure boundary is judged not to be more than a minimal increase. The installation of the 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. 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.

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 "B" 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:** Byron

**Activity/Document Number:** EC 341212 **Revision Number:** 0

**Title:** Installation of Temporary Pumps for Draining the 'B' 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. | |
| √ | 50.59 Evaluation | 50.59 Evaluation No. | 6G-03-0006 | Rev. | 0 |

# 50.59 REVIEW COVERSHEET FORM



**Station:** Byron Unit 2

**Activity/Document Number:** EC 337255 **Revision Number:** 1

**Title:** Containment Floor Drain Sump Flow/Level Instrument Modification and Associated UFSAR DRP, Technical Specification B3.4.15 Bases Change, EC Testing SPP, and Station Procedures

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:

This activity will modify the Containment Floor Drain Leak Detection Instrument Loop (2RF-008) and Containment Floor Drain Sump Level Instrument Loops (2PC-002 and 2PC-003). The following provides a description of existing instrument functional requirements, followed by a description of the proposed modification changes.

Description of Existing 2RF-008 Instrumentation Loop Functions:

The 2RF-008 Instrument Loop functions solely to satisfy Technical Specification 3.4.15, "RCS Leakage Detection Instrumentation". Per UFSAR Section 5.2.5.2.2, this function satisfies RCS Leak-Before-Break (LBB) analysis, with LBB analysis approval based upon meeting conditions of Regulatory Guide 1.45 (Reactor Coolant Pressure Boundary Leakage Detection Systems). Regulatory Guide 1.45 recommends that flow rates of unidentified RCS leakage be monitored and should be capable of detecting 1 gpm leakage increase in the normal flow rates in 1 hour or less. To meet Regulatory Guide 1.45, the 2RF-008 loop provides a High Flowrate Alarm (set for 1 gpm) and flowrate indication on a Digital Chart Recorder. 2RF-008 determines flowrate based upon the relationship of level behind a Weir Plate located within the "unidentified" leakage weir box, which is mounted on the wall of the containment floor drain sump. The RCS leakage detection requirement is satisfied by either the High Flowrate Alarm or through hourly operator monitoring of flowrate indications. Tech Spec 3.4.15 requires only "one containment sump monitor" operable (2RF-008, 2PC-002, or 2PC-003) and "one containment atmosphere radioactivity monitor" operable. Per UFSAR Appendix A, page A1.45-1 (Regulatory Guide 1.45 Clarification), the 2RF-008 Loop *"is designed to remain functional after a Safe Shutdown Earthquake (SSE) and is powered by non-ESF buses."* The existing 2RF-008 and its power supply do not meet the criteria of UFSAR Section 3.2.1.1 for Safety Category I functions. Therefore, the power supply for containment floor drain sump flow channel (2RF-008) is not required to be classified safety related. These existing design conditions will also apply to the new transmitter and recorder being installed in this modification, since the new transmitter and Digital Chart Recorder are being installed as seismically qualified but powered by a non-ESF power supply. In addition, the existing 2RF-008 loop function is not designed to be single failure proof since it consists of a single channel, in which a single failure of any loop component will cause loss of 2RF-008 function. Therefore, a single failure in the existing 2RF-008 will result in loss of alarm function. If this single instrument loop fails, Tech Spec 3.4.15 Bases permits use of the 2PC-002 or 2PC-003 Loops for detecting unidentified leakage of 1 gpm within one hour based on the relationship of floor drain sump level change over time (instead of the Weir Plate relationship). Therefore, the 2PC-002 and 2PC-003 instruments currently are approved for use to satisfy RCS Leakage detection requirements upon failure or loss of the 2RF-008 instrument function.

Description of Existing 2PC-002 and 2PC-003 Instrumentation Loops Functions:

The 2PC-002/003 Loops (Redundant Containment Floor Drain Sump Level Loops) function to satisfy Technical Requirements Manual (TRM) 3.3.i, "Post Accident Monitoring (PAM) Instrumentation". This instrumentation is described in UFSAR Section 6.2.1.7 and UFSAR Appendix E. This level instrumentation is designed Safety Related, Seismic Category I, Environmentally Qualified, and utilizes ESF power supplies. Therefore, at least one train will remain available to monitor sump level following a design basis seismic event, loss of a single train of ESF power, single failure in either instrument loop, etc. In addition to PAM functions, these instrument loops function to satisfy the RCS Leakage Detection Function of Tech Spec 3.4.15. However, these instrument loops cannot currently directly calculate and display flowrate (except through Process Computer PI OILS Display) and do not contain a High Flowrate Alarm function capable of being set at 1 gpm in the same manner as the 2RF-008 Loop. Since 2PC-002 and 2PC-003 currently do not have alarm capability, operators must monitor 2PC-002 or 2PC-003 at least every hour to assure meeting the function of detecting a 1gpm RCS leak within one hour. These instruments are currently being used for RCS Leakage Detection on Unit 2 due to 2RF-008 instrument failure.

# 50.59 REVIEW COVERSHEET FORM

**Station:** Byron Unit 2

**Activity/Document Number:** EC 337255 **Revision Number:** 1

**Title:** Containment Floor Drain Sump Flow/Level Instrument Modification and Associated UFSAR DRP, Technical Specification B3.4.15 Bases Change, EC Testing SPP, and Station Procedures

Description of Modification to 2RF-008 and 2PC-002/003 Instrument Loops:

This modification will disable/abandon the existing "weir plate" method of determining RCS leakage. The 2RF-008 RCS Leakage Detection function (flowrate indication and high flowrate alarm) will be modified by relocating and extending the RCS Leakage Detection bubbler tube into the Containment Floor Drain Sump to provide automatic sump flowrate indication and high flow alarm functions based on floor drain sump level change over time (instead of level behind Weir Plate). Additionally, the existing Containment Floor Drain leak detection loop transmitter (2FT-RF008) will be replaced with a wider range transmitter to measure the wider span of the sump compared to the smaller span of the weir box. The new bubbler tube/transmitter configuration will perform the same function and fulfill the requirements of Technical Specification 3.4.15, "RCS Leakage Detection Instrumentation".

The existing Containment Floor Drain Sump Level Transmitters (2LT-PC002 and 2LT-PC003) will be replaced. The existing Differential Pressure type level transmitters and associated bellows and liquid filled sensing lines will be replaced by two new GEMS Corporation float type resistance level transmitters/probes and associated signal conditioners. The new float level probes will be mounted in a bracket on the containment floor elevation (377'), penetrate the floor drain sump cover, penetrate existing 2RF-008 weir box, and will be rigidly mounted to a bracket on the wall of the floor drain sump. The Post Accident Monitoring function and requirements of 2PC-002/003 to satisfy Technical Requirements Manual (TRM) 3.3.i, "Post Accident Monitoring (PAM) Instrumentation" are not being changed or altered.

The following describes the proposed modification in more detail:

Tech Spec 3.4.15 Bases permits use of the 2PC-002 or 2PC-003 Loops for detecting unidentified leakage of 1 gpm within one hour based on the relationship of floor drain sump level change over time. Therefore, the 2PC-002 and 2PC-003 instruments currently are approved for use to satisfy RCS Leakage detection upon failure or loss of the 2RF-008 instrument function. To provide leak detection flowrate and alarm indication via the modified 2RF-008 instrument loop, in addition to flowrate and alarm indication via the modified 2PC-002/003 instrument loops, the existing Main Control Room Digital Chart Recorder (2FR-RF008) will be replaced with a different model Digital Chart Recorder capable of performing the needed mathematical functions to determine flowrate properly and provide necessary alarm actuations.

The new chart recorder will be programmed to directly calculate flowrate, provide flowrate indication, and provide a high flowrate alarm based on level input from 2RF-008. Containment floor drain sump leakage flow based on level change is calculated utilizing a conversion of 18.23 gallons/inch, in accordance with procedure 2BOSR RF-1. This conversion is based on the physical size of the RF sump, neglecting internal components (i.e., pumps, piping, etc.). This is a conservative approach, as inclusion of internal sump components would decrease the total volume available, resulting in a correction factor less than 18.23 gallons/inch, and, in turn, decreasing the calculated flow rate for an equivalent level change. Outputs from the Digital Chart Recorder will be wired into the existing 2RF-008 MCR Annunciator box to provide both a high flowrate alarm (set at 1 gpm) and a sump high/low level alarm (set just above/below normal sump level) to alert operators of abnormal sump level conditions (note, the new sump high/low level alarms are being added to alert operators of potentially abnormal sump pump operation, which could impact sump flowrate calculations and response time, as described in this evaluation). The new chart recorder will be programmed to prevent faulty flowrate indications during the level transient during times of normal Containment Floor Drain Sump pump operation. When the sump pump operates (normal condition), sump level is decreased rapidly, which will result in loss of steady state sump flowrate indication. After the sump pumpdown "transient" is complete, it may take a period of time for steady state flowrate indication to return to its pre-pump down value. The chart recorder will be programmed to lock out the high flow alarm actuation until steady state flowrate indication can return after sump pump down. This alarm lock out function will be evaluated for a maximum of 30 minutes for the 2RF-008 alarm and 15 minutes for the 2PC-002/003 alarm, with actual settings made as short as practical based on field testing necessary to eliminate unnecessary alarms or incorrect flow indications. The impact of the alarm lockout times has been found acceptable in meeting overall RCS Leakage Detection response time requirements of 1 hour or less, when including other additional increases in response time created from the new level transmitters (both 2RF-008 and 2PC-002/003). Actual overall instrument loop response time and accuracy to flowrate will be verified acceptable through modification testing to actually demonstrate that a 1 gpm leak will be detected within 1 hour upon reaching the sump by both 2RF-008 and 2PC-002/003.

**Station:** Byron Unit 2

**Activity/Document Number:** EC 337255 **Revision Number:** 1

**Title:** Containment Floor Drain Sump Flow/Level Instrument Modification and Associated UFSAR DRP, Technical Specification B3.4.15 Bases Change, EC Testing SPP, and Station Procedures

Level Loops 2PC-002 & 2PC-003 are also available to be used as a backup to satisfy RCS Leakage detection upon failure or loss of the 2RF-008 instrument loop function. As such, the new chart recorder will also be programmed to directly calculate flowrate and provide a high flowrate alarm based on the time elapsed between level steps of the float passing the reed switches spaced at ½" intervals within the new 2PC-002/003 float level probes. Flowrate indication and calculations based on the new 2PC-002/003 loops will normally be "dialed out" on the recorder, but is available if the normal 2RF-008 instrument function is lost. Therefore, if the 2RF-008 loop (primary instrument for leak detection monitoring) is lost for any reason, flow indication from the 2PC-002/003 will be activated and used for RCS leakage detection.

Additionally, this design change will provide a vacuum breaker (opening) in the piping between the Floor Drain Sump and the oil separator area (in response to CR 178722) and will route new 3-conductor cables for existing Containment Floor Water level (wide range) transmitters 2LT-PC006 and 2LT-PC007 from the associated local junction box to the containment electrical penetration. The 2RF-008 Weir Box shall be abandoned and inlet piping rerouted. A metal plate is being installed between the 2RF-008 and 2RF-009 Weir Boxes. This plate will shield the new level probes from potential water overflow from the 2RF-009 Weir Box into the 2RF-008 Weir Box in the event a large influx of water into the 2RF-009 Weir Box (which may occur during RCDT discharge).

In addition to improving RCS Leakage Detection, an additional improvement to existing instrumentation can be made as a result of this modification. Due to the 6" larger level span of the new GEMS float type probes, this new system can be used to enhance operator indication of water level above the containment floor elevation of 377'. The new sump level span will extend 16" above the top of the sump at the 377' elevation, instead of existing 6" above the top of the sump for the existing 2PC-002/003. This instrument enhancement, which is available only due to this mod, will make use of an existing spare 7300 system alarm card currently installed but not used in the 2PC-002 and 2PC-003 instrument loops (2LSH-PC002 and 2LSH-PC003). This alarm card will now be used to provide input to the Containment Sump Level light box (5th light of 2LL-SI075A and 2LL-SI075B), which will provide another redundant indication for verifying minimum post LOCA Containment Floor Water Level prior to isolating the Refueling Water Storage Tank (RWST) from the RH pumps. Currently, this function is acceptably performed by the Containment Floor Water Level Instruments (2PC-006 and 2PC-007), however, the proposed enhancement will improve overall accuracy and human factoring of this function. This enhancement will be installed under this modification, but not be utilized by operators until the applicable Emergency Operating Procedures (EOPs) are revised and other requirements are met to reflect usage of this new instrument capability. The UFSAR, Technical Specifications, and station procedures and design documents currently do not describe the function of the 5th light of 2LL-SI075A and 2LL-SI075B. Therefore, this enhancement will have no impact on plant operations between the time its installed and utilized. A separate Engineering Change activity will be performed to formally evaluate using this enhanced capability for Containment Floor Water Level indication within the Emergency Operating Procedures.

## Reason for Activity:

This change was requested due to repeated loss of function of the existing RCS Leakage Detection Instrumentation. The existing 2RF-008 loop has been declared inoperable numerous times during the current and past operating cycles due to blockage of the system Weir Plate, especially when non-RCS leakage continuously flows through the Weir Box. Blockage (caused by secondary system leakage) results in level behind the Weir Plate to increase higher than it would in a "clean" system for the same actual flowrate. This results in actuating the 1 gpm High Flowrate alarm, although actual flowrate into the Weir Box is less than 1 gpm. Since the Weir Plate is located within the floor drain sump inside the containment missile barrier, it cannot be cleaned without plant shutdown. With the High Flowrate Alarm in solid, its function is declared inoperable. With the alarm inoperable, operators are required to check for RCS leakage via the 2PC-002 or 2PC-003 every hour in order to satisfy 1 hour response time requirements for RCS Leakage Detection. While in this condition, the 2PC-002 and/or 2PC-003 transmitters have also been declared inoperable at various times due to degraded transmitter response time caused by air entrainment within the fluid filled dP level transmitter sensing lines. At times, these failures have left only a single channel of RCS Leakage Detection "sump monitors" operable to satisfy Tech Spec 3.4.15 requirements at various times. The proposed modification was requested to resolve the recurring problems and failures with the existing instrumentation. The new bubbler tube/transmitter configuration is not prone to the same failure mode experienced for the existing 2RF-008 system, as there will no longer be a Weir Box to plug. The sump instrumentation is currently a station Top 10 Material Condition Issue.

**Station:** Byron Unit 2

**Activity/Document Number:** EC 337255 **Revision Number:** 1

**Title:** Containment Floor Drain Sump Flow/Level Instrument Modification and Associated UFSAR DRP, Technical Specification B3.4.15 Bases Change, EC Testing SPP, and Station Procedures

The new GEMS float type level probes are not prone to the same failure modes experienced for the existing 2PC-002/003 systems. There will no longer be long dP level transmitter sensing lines to acquire air. Although the new float level probes have different potential failure modes, the probability and frequency of these failure modes has been determined less than the existing instrumentation. The new float type level probe has been qualified as required and is used in similar applications at other nuclear power facilities. In addition, the new level probes will not be impacted by changes in containment pressure in the manner of the existing instrumentation. The existing 2PC-002/003 indication may shift slightly during changes in containment pressure, which is inherent to the design of the differential pressure transmitters with long fluid filled sensing lines. The new level probes will provide a more stable indication and eliminate these small indication changes which may be distracting during normal containment pressure changes/releases.

The following condition identified in CR 178722 is also being addressed with the modification, "*For Unit 1, during B1R12, following installation of the temporary sump pump, water was still flowing into the RF sump. Investigation determined that when the temporary pump allowed overflow of the oil separator, the RF sump filled to within approximately 2-3 feet from the top. The condition filled the pipe between the oil separator and the RF sump. The completely filled pipe "hydraulically locked," therefore any inventory in the oil separator caused transfer of water to the RF sump. Pumping down the oil separator to the point where air was allowed into the transfer pipe, which broke the siphon, alleviated this condition.* As such, to prevent recurrence of this condition, a corrective action was assigned to include a 'vacuum breaker' in the GEMs modification designs (EC #337254 & 337255). Since this modification already requires re-routing of this piping, a vacuum breaker (consisting of a simple opening/hole) will be added to the piping to prevent hydraulic locking between oil separator location and the floor drain sump.

The existing 2-conductor cable for existing containment water level (wide range) transmitters (2LT-PC006 and 2LT-PC007) is routed in the same conduit as that for the existing 2LT-PC002 and 2LT-PC003 level transmitters and must be removed in order to route new 3-conductor cable for the new GEMS float level transmitters/probes. Therefore, new 3-conductor cable for existing transmitters 2LT-PC006 and 2LT-PC007 will be routed as a contingency if these transmitters are replaced in the future with similar GEMS style level instrumentation.

## Effect of Activity:

The RCS Leakage Detection Instrumentation requirements of Technical Specification 3.4.15 are based upon Regulatory Guide 1.45. Per Regulatory Guide 1.45, ""*It is important to be able to associate a signal or indication of a change in the normal operating conditions with a quantitative leakage flow rate. Except for flow rate or level change measurements from tanks, sumps, or pumps, signals from other leakage detection systems do not provide information readily convertible to a common denominator. Approximate relationships, which convert these signals to units of water flow, should be formulated to assist the operator in interpreting signals. Since operating conditions may influence some of the conversion procedures, the procedures should be revised during such periods.*" The activities in this EC are consistent with Reg. Guide 1.45, UFSAR Appendix A, and Technical Specification Bases 3.4.15 since a known relationship between sump level and flowrate is being utilized. This known relationship will be substituted for the known relationship between 2RF-008 Weir Box level versus flowrate currently utilized. Therefore, the use of the new bubbler tube/transmitter configuration in this manner is consistent and essentially the same as currently approved leakage detection methodology. The revised instrument scaling relationship will readily allow converting level indication into the desired common denominator (flowrate). Although the response time of the new leakage detection systems are increased with this modification, the overall requirements of detecting 1 gpm within 1 hour will continue to be met.

The relationship of sump level change over time versus flowrate will be confirmed acceptable, including allowances for instrument uncertainty and obstructions in the sump which could affect the sump volume (i.e., sump pumps, pipes, etc.). The accuracy and required response time will be tested in the field prior to declaring the instrumentation operable. Affects of sump pump down on flow indication and response time will be tested and verified. Operating procedure will be revised to assure the new bubbler tube/transmitter configuration can be utilized in a manner essentially the same as current practice.

The proposed activity will require UFSAR change, as the 2RF-008 Weir Box design is described in detail, to reflect the new configuration for description of leak detection instruments. Additionally, description of Post Accident Monitoring requirements for containment water level will require UFSAR change to reflect the new instrumentation configuration

Station: Byron Unit 2

Activity/Document Number: EC 337255 Revision Number: 1

Title: Containment Floor Drain Sump Flow/Level Instrument Modification and Associated UFSAR DRP, Technical Specification B3.4.15 Bases Change, EC Testing SPP, and Station Procedures

(float-type level transmitters with signal conditioners). However, the new 2RF-008 bubbler tube/transmitter configuration will perform the same function as the existing RCS leak detection loop and the new 2PC-002 and 2PC-003 level instrumentation will perform the same function as the existing Post Accident Monitoring loops. As such, there will be no functional requirement changes as a result of this modification.

With respect to leakage detection instrumentation requirements of Technical Specification 3.4.15, no change to description or function is required. Technical Specification Basis will require change to take credit for leak detection flow alarm function of 2PC-002/003. With respect to Post Accident Monitoring Functions of TRM 3.3.i, no changes to description or function are required.

The existing containment floor drain sump flow channel and its power supply do not meet the criteria of UFSAR Section 3.2.1.1 for Safety Category I structures, systems, and components. Therefore, the power supply for containment floor drain sump flow channel alarm function is not required to be classified safety related. This statement applies to the existing High Flowrate alarm and will apply to the modified High Flowrate alarms, since the Digital Chart Recorder will continue to be powered by a non-ESF power supply. Therefore, there is no change in reliability of alarm function as related to its power supply. In addition, the new Chart Recorder will provide alarm actuation upon loss of power, alerting operators to monitor RCS Leakage via means other than the recorder (i.e., Process Computer PI Indication or MCR Level indicators). This new alarm is in addition to the existing alarm which would alert operators to the loss of the associated non-ESF power supply to the recorder.

The "vacuum breaker" added to the piping between the containment floor drain sump & the oil separator will prevent recurrence under any conditions of this piping being "hydraulically locked" and as a result, prevent any unwanted inventory in the oil separator from being transferred to the RF Floor Drain Sump. Unidentified leakage to the Floor Drain Sump will continue to be monitored by the leakage detection instrumentation and is unaffected by this piping configuration change.

The existing 2PC-006 and 2PC-007 wide range level instrumentation will be evaluated and tested prior to their required use to ensure indication is unaffected by the new 3-conductor cable. After demonstrating its required functionality, the existing 2PC-006 and 2PC-007 wide range level instrumentation will continue perform their intended function.

The instrument enhancement for indication of Containment Floor Water Level above 377' elevation will have no impact to plant operations upon installation of this change. The use of this enhanced function for operator verification of minimum Containment Floor Water Level following a LOCA will not be used immediately (i.e., will be evaluated for use at a future time under separate 50.59 evaluation). The Technical Specifications, UFSAR, Station Design Documents, and procedures have been reviewed to verify this enhancement does not violate any existing requirements. Therefore, this change will be a benefit only after the applicable Emergency Operating Procedures are revised to allow its use in the future.

## Summary of Conclusion for the Activity's 50.59 Review:

The RCS Leakage Detection Instrumentation provides passive indication only and cannot initiate an accident. This instrumentation is installed in a manner so that it does not interact with other SSCs which could initiate or result in creating an accident.

The proposed changes are being implemented to improve instrument reliability over that of the existing design. Therefore, the premise of this modification is to decrease the likelihood of equipment malfunction. The bases for this conclusion are provided.

The consequences of an accident involving a leak in the Reactor Coolant Pressure Boundary as previously evaluated in the UFSAR remains unchanged since the proposed activity does not interact with or support any other SSC used to minimize or mitigate the consequences of a design basis accident. This activity does not alter the function of 2PC-002/003 as required by TRM Section 3.3.i Post Accident Monitoring Instrumentation. The modified Containment Floor Drain Sump Level instrumentation will continue to meet all design requirements.

The consequences of a malfunction which results in failure to detect a 1 gpm RCS leakrate within 1 hour is the same regardless of the instrumentation used to detect the leak. The modified instrumentation remains designed per PAM requirements of Regulatory Guide 1.97 and UFSAR requirements to assure function following an accident and a single

Station: Byron Unit 2

Activity/Document Number: EC 337255 Revision Number: 1

Title: Containment Floor Drain Sump Flow/Level Instrument Modification and Associated UFSAR DRP, Technical Specification B3.4.15 Bases Change, EC Testing SPP, and Station Procedures

failure. The 2PC-002/003 instruments will continue to meet existing design requirements of providing level information to operators and Technical Support Staff as necessary.

Based on the non-intrusive and passive function of the modified instrumentation, the proposed activity will not create the possibility of an accident of a different type than previously evaluated in the UFSAR.

Any malfunction of the modified instrumentation will end in the same result as a malfunction of the existing instrumentation. This result is only loss of passive indication and alarm functions with no impact on other plant functions. Therefore, operation with the modified instrumentation will not create the possibility for a malfunction, which ends in a different result than previously evaluated in the UFSAR.

Design basis limits for a fission product barriers remain unchanged. The modified instrumentation will continue to meet requirements of detecting small amounts of RCS leakage before a potentially larger degradation of the RCPB occurs.

Eliminating use of the Weir Plate methodology for flowrate determination is acceptable since the method for determining flowrate based on sump level change over time is currently allowed per Technical Specification 3.4.15 Bases and Regulatory Guide 1.45 (i.e., both methods of flowrate determination are acceptable and are "essentially the same").

No Technical Specification or Operating License change is required for this activity. However, Technical Specification Basis will require change to take credit for leak detection flow alarm function of 2PC-002/003.

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-04-0002 | Rev. | 1 |

| Procedure Revisions and Special Process Procedures (SPP) | |
|---|---|
| 1. | 6H-01-0009 |
| 2. | 6G-02-0015 |
| 3. | 6G-03-0001, Revision 1 |
| 4. | 6G-03-0002 |

Safety Evaluation Summary Form

Tracking No. 6H-01-0009
Activity No. BOP SX-M1, Revision 24

DESCRIPTION:
Revise Byron Operating Procedure (BOP) SX-M1 to change valve lineup positions of 0SX161A/B to "Open" instead of "Throttled" due to incorporation of restricting flow orifices. Revise procedure to add valves 0SX251A/B and 0SX252A/B to the valve lineup sheet. These instrumentation valves are installed for periodic testing of the blowdown flow and are identified in the "Closed" position. Additional minor changes facilitate procedure enhancement and performance.

SAFETY EVALUATION SUMMARY:

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

   The probability of occurrence or the consequence of an accident or a malfunction of equipment important to safety previously evaluated in the SAR is not increased because the flow orifices have been tested to verify the UFSAR limit of 300 gpm of blowdown flow is not exceeded. This allows 0SX161A/B to be fully opened, which is the change identified in BOP SX-M1. Valves 0SX251A/B and 0SX252A/B are installed in the flow element high and lo side flanges for periodic testing of the blowdown lines. These are maintained in the "Closed" positions until required for testing.

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

   The possibility of an accident or malfunction of a different type is not increased because the UFSAR is being updated per DRP 7-258 to identify the flow restricting orifices and deleting the requirements of limiting flow through valve positioning. BOP SX-M1 is a reference document to support configuration control activities and establishes positioning of 0SX161A/B in accordance with previously evaluated accidents or malfunctions.

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

   The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the identified criteria as implemented in the proposed changes of the procedure are in accordance with the guidelines and requirements of the UFSAR.

# 50.59 REVIEW COVERSHEET FORM

**Page 1 of 1**

**Station: Byron**

**Activity/Document Number** SPP 02-015 and EC 340013 **Revision Number:** 0/0

**Title:** Special Procedure for Feedwater Loop Data Collection

**Description of Activity:**

The proposed activity will manually manipulate an individual Feedwater Regulating Valve (FRV) to raise or lower Steam Generator water levels to create an error signal between program Steam Generator level and actual Steam Generator level. The FRV will than be placed in automatic control to observe the Steam Generator Water Level Control system response as it restores the actual Steam Generator level to programmed level. Limitations and Actions section of the proposed activity require a downpower to 97% Reactor power to avoid exceeding licensed thermal power limits. The proposed activity also requires that the standby CD/CB pump be run to offset any decrease in FW pump suction pressure that may occur when a FRV responds to low Steam Generator level and opens to increase FW pump flow demands.

**Reason for Activity:**

The purpose of this activity is to collect data from the steam generator water level control system. Utilizing data-gathering equipment from Entech control, the FRV for an individual Steam Generator will be manipulated in automatic and manual modes to observe control parameters such as process gain, dead band, time delays, response time and valve positioning. Also the interactive responses of the FW Pump Speed Control System and FRV control will be observed during manual manipulation of the FRV and level changes in automatic mode. The information gathered will be used to determine if changes to controller settings and/or control instrumentation are required.

**Effect of Activity:**

Limitations and Actions section of the proposed activity require a downpower to 97% Reactor power to avoid exceeding licensed thermal power limits. The proposed activity also requires that the standby CD/CB pump be run to offset any decrease in FW pump suction pressure that may occur when a FRV responds to low Steam Generator level and opens to increase FW pump flow demands. This will prevent receipt of a FW pump NPSH low alarm and its associated automatic actions. The proposed activity directs operators to start back up pressurizer heaters to establish pressurizer spray valve flow demand. This will provide for a more rapid response to any primary pressure transients and mitigate any affects that may be induced by changes in feedwater flow. Based on the above required limitations and the fact that the proposed activity operates the feedwater system with in the guidelines set fourth in UFSAR Section 13.5.2, Operating and Maintenance procedures, there will be no adverse effects on any SSC from this activity.

**Summary of Conclusion for the Activities 50.59 Review:**

The proposed activity will have no adverse impact on UFSAR described design functions or design basis limits. The change does not involve a change to a procedure that adversely affects UFSAR described design functions. The change does not revise or replace UFSAR described evaluation methodology. The change does not involve a test or experiment where an SSC is utilized or controlled outside the bounds of the existing design or inconsistently with UFSAR analysis or descriptions. The change has no impact on Technical Specifications or Operating License. No new SSC failure modes are created.

**Attachments:**
Attach completed Applicability Review if 50.59 Screening is not required.
Attach completed 50.59 Screening if 50.59 Evaluation is not required.
Attach completed 50.59 Evaluation if required to be performed.
Attach completed 50.59 Screening and 50.59 Evaluation if multiple discrete elements of an activity have been linked together and certain elements required a 50.59 Evaluation while other elements did not.

**Forms Attached:** (Check all that apply.)

| | | | | | |
|---|---|---|---|---|---|
| ☐ | **Applicability Review** | | | | |
| ☐ | **50.59 Screening** | **50.59 Screening No.** | | **Rev.** | |
| ☒ | **50.59 Evaluation** | **50.59 Evaluation No.** | 6G-02-015 | **Rev.** | 0 |
| ☐ | **50.59 Validation** | **50.59 Validation No.** | | **Rev.** | |

# 50.59 REVIEW COVERSHEET FORM



**Station:** *Byron/Braidwood*

**Activity/Document Number:** 1/2BEP/BwEP-0, 1/2BEP/BwEP-1, 1/2BEP/BwEP ES-1.2, 1/2BEP/BwEP ES-1.3

**Revision Number:** See below

**Title:** *Revision of BEP/BwEP-0, -1, ES-1.2, ES-1.3, to Implement Changes to SVAG Valve Reenergization*

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.)

*NOTE: Rev. 0 of this 50.59 Evaluation was prepared and reviewed in early April 2003, but was not used at that time because the procedure changes were deemed to not be required. Subsequently, in late May 2003, it was determined that the procedure changes would be required. This revision updates the revision numbers of 1/2BEP-0 and 1/2BEP ES-1.3 to account for other procedure revisions that occurred between April and May 2003.*

*This evaluates the following procedure revisions:*

1. *Byron Procedures 1/2BEP-0, Reactor Trip or Safety Injection, Unit 1/2, rev. 104*
2. *Byron Procedures 1/2BEP-1, Loss of Reactor or Secondary Coolant, Unit 1/2, Rev. 103*
3. *Byron Procedures 1/2BEP ES-1.2, Post LOCA Cooldown and Depressurization, Unit 1/2, Rev. 102*
4. *Byron Procedures 1/2BEP ES-1.3, Transfer to Cold Leg Recirculation, Unit 1/2 Rev. 100, Interim change 03-1-038/101, Interim change 03-2-038*
5. *Braidwood Procedures 1/2BwEP-0, Reactor Trip or Safety Injection, Unit 1/2 Rev. 101/103*
6. *Braidwood Procedures 1/2BwEP-1, Loss of Reactor or Secondary Coolant, Unit 1/2, Rev. 104/103*
7. *Braidwood Procedures 1/2BwEP ES-1.2, Post LOCA Cooldown and Depressurization, Unit 1/2, Rev. 102*
8. *Braidwood Procedures 1/2BwEP ES-1.3, Transfer to Cold Leg Recirculation, Unit 1/2 Rev. 101*

*These procedures are being revised to provide operator actions, prior to reaching automatic switchover to Cold Leg Recirculation, to deenergize the Spurious Valve Actuation Group (SVAG) valve Motor Control Centers (MCCs) using the Main Control Room (MCR) switches and to dispatch an operator to the SVAG valve MCCs to close in the breakers locally; and steps to reenergize the SVAG valve MCCs using the MCR switches when required to reposition the SVAG valves after switchover to Cold Leg Recirculation. These actions replace the current actions, which dispatch operators to locally close the breakers later in the event, just prior to the steps that reposition the SVAG valves.*

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

*The capability to implement the current method of locking out power locally to the SVAG valves has been questioned. This is because potentially excessive radiation dose rates at the local MCCs, at the time the procedures currently require the breakers to be energized could prohibit access to the areas.*

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

*Swapping the power lockout from the local breakers to the MCR-operated MCC feeder breakers prior to transfer to Cold Leg Recirculation will maintain the required power lockout, but change the timing of the closure of the local breakers to a point where the expected radiation dose will be acceptable. This is a compensatory action in support of Byron Op Eval 03-003 and Brwd Op Eval 03-002. Because this is a compensatory action in accordance with GL 91-18 to deal with a degraded condition, this 50.59 evaluation is being done to "determine whether the compensatory action itself (not the degraded condition) impacts other aspects of the facility described in the UFSAR."*

# 50.59 REVIEW COVERSHEET FORM

**Station:** *Byron/Braidwood*

**Activity/Document Number:** 1/2BEP/BwEP-0, 1/2BEP/BwEP-1, 1/2BEP/BwEP ES-1.2, 1/2BEP/BwEP ES-1.3

**Revision Number:** See below

**Title:** *Revision of BEP/BwEP-0, -1, ES-1.2, ES-1.3, to Implement Changes to SVAG Valve Reenergization*

**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 activity proposes changes to Emergency Procedures that are used in response to a Loss of Coolant Accident. The steps in the Emergency Procedures are not initiators of any accidents. Therefore, these changes cannot increase the frequency of occurrence of any accidents previously evaluated in the UFSAR. The proposed procedure changes return the operation of the SVAG valves in compliance with the description in UFSAR section 8.1.10 as related to single failure criterion and power lockout to motor operated valves, as reviewed by the NRC in response to FSAR Question 040.162 and approved in the Byron SER (NUREG-0876) and the Braidwood SER (NUREG 1002), Section 8.4.3, which meets the provisions of Branch Technical Position ICSB 18 (PSB) "Application of the Single Failure Criterion to Manually-Controlled Electrically Operated Valves". The likelihood of occurrence of a failure of the MCC feeder breakers is unchanged. Because no physical changes are being made to any plant equipment and because the preventive testing of the breakers is unchanged, there is no increase in the likelihood of occurrence of a malfunction of any SSC important to safety. The proposed activity does not result in any increase in the consequences of a LOCA, nor do they affect the performance of the ECCS as supported by the operation of the SVAG valves as the plant operators will be able to operate the SVAG valves as necessary to support the assumptions of the accident analysis. The SVAG valves were de-energized prior to this change and will remain de-energized until their operation is necessary as identified in the Emergency Procedures. The proposed procedures changes do not result in changing the failure mode of the non-SVAG valves. A failure analysis for these valves is included in UFSAR Table 6.3-10 and is not affected by these changes. This activity does not introduce the possibility of a change in the consequences of a malfunction because changing the steps in the Emergency Procedures is not an initiator of any new malfunctions and new failure modes are not introduced. The proposed configuration will have the breakers for the SVAG valves at the local Motor Control Centers (MCC) closed (racked in) with power to the MCCs removed via opening the control switches for the 480 volts bus feeds at the Main Control Boards. This change re-aligns the operation of the SVAG valves with the description given in UFSAR section 8.1.10.*

*There are a number of other valves that are powered from the same MCCs that feed the SVAG valves. Prior to the proposed procedures changes, these valves were energized at all times. Due to the proposed changes, these valves will be de-energized when the main control room switches for the feed to the affected 480 volts busses are taken to the "Trip" position. These valves will be re-energized when these switches are closed later in the event. However, there is a change to the way electrical power to the valves is managed. Table 6.3-10 of the Byron and Braidwood UFSAR provides a Failure Mode and Effects Analysis for the ECCS active components. These valves will continue to meet the analysis of Table 6.3-10, as alternate isolation is available. In addition, failure of one of these valves to close would result in draining RWST inventory to the Containment. This additional outflow from the RWST has been considered in the evaluation of the RWST minimum drawdown time; this feeds into the evaluation of the minimum time available for the operator to complete the ECCS switchover to the RCS Cold Legs sequence. Therefore, the failure of any one of valves does not result in a minimal increase of the consequences of a malfunction of an SSC important to safety.*

*This activity does not create the possibility of an accident of a different type because none of the procedure steps are initiators of an accident.*

*All of the components manipulated by these procedure steps were designed for this activity. The worst case malfunction is that a breaker may not close or open upon demand. These procedure steps do not change those malfunctions. The steps taken prior to switchover to Cold Leg Recirculation put the unit into the configuration already reviewed and approved by NRC in the Byron SER (NUREG-0876) and Braidwood SER (NUREG 1002), Section 8.4.3. Failure of the non-SVAG valves that will be temporarily de-energized has been evaluated in UFSAR Table 6.3-10; the failure mode is not changed and the result of the failure is not changed for any of these valves.*

# 50.59 REVIEW COVERSHEET FORM



**Station:** *Byron/Braidwood*

**Activity/Document Number:** 1/2BEP/BwEP-0, 1/2BEP/BwEP-1, 1/2BEP/BwEP ES-1.2, 1/2BEP/BwEP ES-1.3
**Revision Number:** See below

**Title:** *Revision of BEP/BwEP-0, -1, ES-1.2, ES-1.3, to Implement Changes to SVAG Valve Reenergization*

*The procedure changes have been simulator-validated to ensure that there is no adverse impact on required operator action times such as completion of Cold Leg Switchover, SG Tube Rupture Margin To Overfill, etc.).*

*The proposed procedure changes do not impact any design basis limit for a fission product barrier. These steps ensure that the SVAG valves remain locked out in the short term (until switchover to Cold Leg Recirculation) while still allowing them to be reenergized and stroked as necessary. Therefore, the ECCS will operate as designed to provide core cooling. In addition, the proposed changes do not impact the operation of other Engineered Safety Feature equipment so that there is no impact on any design basis limit for the fuel cladding, RCS or containment.*

*The revised procedures will support transfer of the ECCS to the Cold Leg Recirculation and to the Hot Leg Recirculation mode of operation. The changes do not affect the sequence of ECCS operation assumed in the safety analyses. Thus, the procedure changes do not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.*

**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.** | *Byr 6G-03-0001*<br>*Brw-E-2003-93* | **Rev.** | *1* |

# 50.59 REVIEW COVERSHEET FORM

LS-AA-104-1001
Revision 1

50.59 Evaluation No.: BRW-E-2003-94 / 6G-03-0002 Rev. No.: 0, 0 Page 1 of 2

Station: Braidwood/Byron Units 1 and 2

Activity/Document Number: 1(2)BwEP-0 and 1(2)BEP 0 Revision Number: 101 (1BWEP-0), 103 (rest)

Title: Revision to Emergency Procedures to provide for manual shutdown of VV (Byron and Braidwood), VL (Braidwood) and VW (Braidwood) systems following a Safety Injection actuation

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:**

The Braidwood procedures are being revised to provide directions to shutdown the Miscellaneous/Shift Office ventilation (VV), Laboratory ventilation (VL) and Radwaste Building ventilation (VW) systems and the Byron procedures are being revised to provide directions to shutdown the Miscellaneous/Shift Office ventilation (VV) following a Safety Injection signal.

**Reason for Activity:**

Due to the physical arrangement of VV, VL, VW and Control Room Ventilation (VC) ducting, a potential for unfiltered leakage into the VC system exists from leakage out of the VV, VL and VW systems. To address this, duct leakage testing was performed on the VL and VW systems at Byron during construction eliminating the need to shut down these two systems. In lieu of completing the duct leakage testing, Braidwood initiated shut down of the VL and VW systems for high outside air radiation detected by the VC emergency filtration actuation radiation monitors (0PR31J, 32, 33J and 34J). Neither station performed leakage testing of the VV system, opting to shut down the system on high outside radiation.

Following a Safety Injection signal, the VC system will realign for Emergency Makeup Mode operation and the source of makeup air is automatically realigned from the outside air intake to the Turbine Building air intake. When the VC system makeup air is aligned to the Turbine Building air intake, there is no longer forced outside air flow past the outside air radiation monitors and they may not be capable of effectively monitoring outside air radiation levels (ref. CR 141389 (Braidwood) and 141542 (Byron)). To address this concern, the subject emergency procedures are being revised to preemptively shut down the VV system at Byron and the VV, VL and VW systems at Braidwood following a Safety Injection signal.

**Effect of Activity:**

The VV, VL and VW ventilation systems do not provide ventilation to areas of the plant or plant equipment required following a Safety Injection actuation. The procedure change will align these systems in a configuration to obtain control room ventilation isolation following a safety injection actuation prior to any potential radiation release. The procedure change will add manual actions to the existing sequence of events for the emergency as part of the sub-steps associated with verifying the control room ventilation system is aligned for emergency operation. This sub-step is typically done in parallel with other steps in the procedure. Based on this, changing the action for tripping the ventilation fans from a Response Not Obtained action to a direct action would not affect the overall time to perform the procedure.

**Summary of Conclusion for the Activity's 50.59 Review:**

The procedure change does not result in a change to the frequency of occurrence of any accident or failure of any SSC important to safety. The consequences of any accident or failure of an SSC important to safety are not changed. No new accidents or failures of SSCs are introduced as a result of the procedure change. The changes to the procedures do not adversely affect how UFSAR described SSC design functions are performed. No changes in evaluation methodologies were required as the result of the procedure revisions. No changes to Technical Specifications are required. Therefore, the changes to the procedure can be made without prior NRC approval.

# 50.59 REVIEW COVERSHEET FORM

LS-AA-104-1001
Revision 1

**50.59 Evaluation No.:** BRW-E-2003-94 / 6G-03-0002 **Rev. No.:** 0, 0 **Page 2 of 2**

**Station:** Braidwood/Byron Units 1 and 2

**Activity/Document Number:** 1(2)BwEP-0 and 1(2)BEP 0 **Revision Number:** 101 (1BWEP-0), 103 (rest)

**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.)

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|---|---|---|---|---|---|
| | **Applicability Review** | | | | |
| | **50.59 Screening** | **50.59 Screening No.** | | **Rev.** | |
| X | **50.59 Evaluation** | **50.59 Evaluation No.** | BRW-E-2003-94 / 6G-03-0002 | **Rev.** | 0, 0 |

| Technical Requirements Manual (TRM) Changes and Draft Revision Packages (DRP) | |
|---|---|
| 1. | 6G-03-0004 |
| 2. | 6G-03-0005 |
| 3. | 6G-03-0008 |
| 4. | 6G-04-0001 |

# 50.59 REVIEW COVERSHEET FORM



**Station:** *Byron*

**Activity/Document Number:** *TRM Change 03-013* **Revision Number:** *N/A*

**Title:** *TRM Change 03-013, TRM 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 B1R12 from 100 hours to 56 hours:*

*Byron TRM Section 3.9.a, "Decay time," states "The reactor shall be subcritical for ≥ the last 100 hours (≥ 57 hours for B2R10)." This activity will revise this statement to replace "(≥ 57 hours for B2R10)" with "(≥ 56 hours for B1R12)."*

*Byron TRM Section 3.9.a, Action Condition A states "Reactor subcritical for < 100 hours (< 57 hours for B2R10)." This activity will revise this statement to replace "(< 57 hours for B2R10)" with "(< 56 hours for B1R12)."*

*Byron TRM Surveillance TSR 3.9.a.1 states "Verify the reactor subcritical ≥ 100 hours by confirming the date and time of subcriticality. (≥ 57 hours for B2R10)." This activity will change "(≥ 57 hours for B2R10)." to "(≥ 56 hours for B1R12)."*

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

*It is anticipated that during B1R12, 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 spent fuel pool cooling analysis assumes that fuel transfer begins after 100 hours decay time in the reactor core. This evaluation is thus required to determine if the proposed changes can be made under the provisions of 10 CFR 50.59.*

*Because past ICDT TRM changes were considered one-time, cycle-specific changes, no UFSAR changes were made. However, since reduction in ICDT is becoming a common practice, UFSAR Section 9.1.3.1 was permanently revised to reflect the option of shorter ICDT's.*

*(Note: This change does not address the radiological consequences of a Fuel Handling Accident (FHA). The radiological consequences of a FHA have been 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 changes will allow starting B1R12 fuel offloading activities earlier than the current 100 hours. This will save time on the critical path for the outages. Occupational dose on the refueling machine may increase slightly.*

# 50.59 REVIEW COVERSHEET FORM



**Station:** *Byron*

**Activity/Document Number:** *TRM Change 03-013* **Revision Number:** *N/A*

**Title:** *TRM Change 03-013, TRM 3.9.a, "Decay Time"*

**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 56 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 fuel handling accident due to human error. Spent fuel handling tools will not change, nor will the method/procedures for handling spent fuel assemblies. The total number of fuel assemblies to be transferred, and the transfer rate, remains the same. There is no effect on the failure probabilities of the Spent Fuel Pool cooling system.*

*Revision 000E to calculation BRW-00-0010-M/BYR2000-007 has been performed 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 this analysis show that the maximum bulk water temperature calculated in the design basis, and the time-to-boil evaluation, are not altered by changing the ICDT from 100 hours to 56 hours for B1R12. In addition, it has been verified that the maximum local water temperature, the maximum fuel cladding temperature and the maximum cladding heat flux remain acceptable.*

*The design basis spent fuel pool criticality analysis (for the Spent Fuel 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 spent fuel pool, thus adding negative reactivity. The spent fuel pool criticality analysis is thus 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 56 hrs 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 10CFR20 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 10CFR20 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. Revision 000E to calculation BRW-00-0010-M/BYR2000-007 shows the calculated maximum fuel cladding temperature remains well below the mean cladding operating temperature of 700 °F (far below the design basis limit of 2,200 °F), and the maximum calculated heat flux for a fuel assembly is a fraction of the required heat flux for Departure from Nucleate Boiling (DNB). Therefore no DBLFPB as described in the UFSAR 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 spent fuel pool 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 SER for the Power Uprate Project.*

# 50.59 REVIEW COVERSHEET FORM



**Station:** *Byron*

**Activity/Document Number:** *TRM Change 03-013* **Revision Number:** *N/A*

**Title:** *TRM Change 03-013, TRM 3.9.a, "Decay Time"*

**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-03-0004 | Rev. | 0 |

# 50.59 REVIEW COVERSHEET FORM



**Station:** Braidwood Unit 1/Byron Unit 1

**Activity/Document Number:** UFSAR Update addressing NSAL-03-1/DRP 10-028 **Revision Number:** 0

**Title:** UFSAR Update addressing NSAL-03-1 - Safety Analysis Modeling Loss of Load/Turbine Trip

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.)

DRP 10-028 is prepared to incorporate a change in modeling assumption used in the loss of load/turbine trip (LOL/TT) analysis for Byron and Braidwood Unit 1.

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

Westinghouse Nuclear Safety Advisory Letter (NSAL) 03-1 communicated to licensees of Westinghouse designed PWRs about a modeling issue associated with the LOL/TT analysis. For the LOL/TT analysis peak pressure transient case, Westinghouse typically assumes the initial RCS temperature to be the nominal full power temperature plus the temperature uncertainty. Recent analyses performed by Westinghouse showed that for some plants, subtracting the temperature uncertainty from the nominal full power temperature could delay the actuation of the secondary-side main steam safety valves and result in higher peak RCS pressure. This issue is applicable to Byron and Braidwood. Westinghouse has performed an evaluation to address this issue and determined that the peak RCS pressure for Byron and Braidwood unit 1 will increase by 41 psi and there is no impact on unit 2. This is due to the replacement steam generators installed at Byron Unit 1 and Braidwood Unit 1. DRP 10-028 updates the UFSAR to reflect the conservative change in modeling and the results associated with the change for Byron and Braidwood unit 1.

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

There is no effect on plant operations.

The effect on the design basis is that the UFSAR will be updated to address the modeling issue identified in NSAL-03-1. The LOL/TT analysis for Byron and Braidwood unit 1, discussed in UFSAR Section 15.2.3, has been revised to assume the initial RCS temperature to be the nominal full power temperature minus the temperature uncertainty. The maximum RCS pressure acceptance criterion continues to be met.

**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.)

DRP 10-028 updates the UFSAR to reflect a conservative change in analysis assumption for Byron and Braidwood unit 1. The activity does not involve a change to an SSC that adversely affects an UFSAR described design function, does not involve a change to a design basis limit for fission product barriers, does not involve a change to a procedure that adversely affects how UFSAR described design functions are performed or controlled, does not require a change to the Technical Specifications or Operating License, and does not result in a departure from a method of evaluation described in the UFSAR used in the safety analysis. The activity may be implemented without prior NRC review and approval.

# 50.59 REVIEW COVERSHEET FORM



**Station:** Braidwood Unit 1/Byron Unit 1

**Activity/Document Number:** UFSAR Update addressing NSAL-03-1/DRP 10-028 **Revision Number:** 0

**Title:** UFSAR Update addressing NSAL-03-1 - Safety Analysis Modeling Loss of Load/Turbine Trip

**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.)

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|---|---|---|---|---|---|
| | **Applicability Review** | | | | |
| ✓ | **50.59 Screening** | **50.59 Screening No.** | | **Rev.** | |
| ✓ | **50.59 Evaluation** | **50.59 Evaluation No.** | BRW-E-2003-148 / 6G-03-0005 | **Rev.** | 0 |

# 50.59 REVIEW COVERSHEET FORM



**Station:** Byron Unit1, Byron Unit 2, Braidwood Unit 1, Braidwood Unit 2

**Activity/Document Number:** UFSAR Update addressing the Revised Structural Component Criterion (Revision 1 to Addendum 1 of WCAP-12488-A) / DRP 10-043

**Revision Number:** 0

**Title:** UFSAR Update addressing the Revised Structural Component Criterion (Revision 1 to Addendum 1 of WCAP-12488-A)

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.)

*

*DRP 10-043 is prepared to incorporate a Fuel Rod Design Method change for the Fuel Structural Hydrogen Content parameter with the implementation of Addendum 1 to WCAP-12488-A. Implementation of Addendum 1 to WCAP 12488-A provides a differentiation between heated (cladding) and unheated (structural) surfaces, and changed the current criteria (below) to the following new criteria (below):*

*

*Current Criteria:* *The hydrogen content of Zircaloy-4 and ZIRLO™ structural components shall be less than 600 ppm*

*

*New Criteria:* *The Zircaloy-4 and ZIRLO™ structural component stresses will be consistent with ASME Code Section III requirements after accounting for thinning due to corrosion.*

*

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

*

*With the approval and implementation of Addendum 1 of WCAP-12488-A, Westinghouse has changed their Fuel Structural Hydrogen Content parameter. In Addendum 1 of WCAP-12488-A, Westinghouse asserted that the current criterion of using a hydrogen pickup limit for structural components is difficult to verify and does not conform to industry guidelines. The NRC staff agreed, with Westinghouse, that structural components, other than cladding, could be analyzed more adequately using criteria based on mechanical properties such as stress, strain, and material strength.*

*

*With Westinghouse's implementation of this new criterion for the upcoming Byron Unit 1 Cycle 13 reload core, Section 4.2 of the UFSAR needs to reflect this change in methodology.*

*

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

*

*There is no effect on plant operations.*

*

*The effect on the design basis is that the UFSAR will be updated to reflect this revised structural component criterion. This UFSAR update will reflect the change from the current NRC approved criterion to the new NRC approved criterion beginning with Byron Unit 1 Cycle 13, Byron Unit 2 Cycle 12, Braidwood Unit 1 Cycle 12, and Braidwood Unit 2 Cycle 11.*

*

*The intent of this criterion is to prevent the loss of ductility due to hydrogen embrittlement by the formation of zirconium hydride platelets. Westinghouse performed analyses on grid strap material and assembly thimble tubes. The analyses confirmed:*

*

*(a) Ductility decreases gradually with increasing hydrogen concentration up to 2000 ppm, and*
*(b) A significant amount of ductility exists at operating temperatures and hydrogen contents up to 2000 ppm.*

*

*The NRC agrees with Westinghouse that structural components, other than cladding, can be analyzed more adequately using criteria based on mechanical properties such as stress, strain, and material strength. This new criterion meets this intent.*

*

# 50.59 REVIEW COVERSHEET FORM



**Station:** Byron Unit1, Byron Unit 2, Braidwood Unit 1, Braidwood Unit 2

**Activity/Document Number:** UFSAR Update addressing the Revised Structural Component Criterion (Revision 1 to Addendum 1 of WCAP-12488-A) / DRP 10-043

**Revision Number:** 0

**Title:** UFSAR Update addressing the Revised Structural Component Criterion (Revision 1 to Addendum 1 of WCAP-12488-A)

**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.)

*

*DRP 10-043 updates the UFSAR to reflect the new Fuel Rod Design criterion for the Fuel Structural Hydrogen Content criterion beginning with Byron Unit 1 Cycle 13, Byron Unit 2 Cycle 12, Braidwood Unit 1 Cycle 12, and Braidwood Unit 2 Cycle 11.*

*

*In WCAP-12488-A, Westinghouse describes a process and criteria that it intends to apply to changes or improvements in existing fuel designs that will not require NRC review and prior approval when these criteria are met. Addendum 1 to WCAP-12488-A revises the Fuel Structural Hydrogen Content parameter criterion to a criterion that can be more adequately analyzed. This new criterion differentiates between heated (cladding) and unheated (structural) surfaces.*

*

*The activity does not involve a change to an SSC that adversely affects an UFSAR described design function. The activity does not involve a change to a design basis limit for fission product barriers. The activity does not involve a change to a procedure that adversely affects how UFSAR described design functions are performed or controlled. The activity does not require a change to the Technical Specifications or Operating License. Lastly, the activity does not result in a departure from a method of evaluation described in the UFSAR for the following reason:*

*

> *The implementation of Addendum 1 to WCAP-12488-A replaces the hydrogen content design criterion for fuel assembly structural components with a structural component stress criterion that accounts for material thinning due to corrosion. The implementation of Addendum 1 to WCAP 12488-A is a change to the fuel evaluation methodology in the UFSAR by reference. As discussed in the SER for Addendum 1 to WCAP 12488-A, the NRC has approved the application of this new criterion for structural components of Westinghouse fuel assemblies. The use of this new criterion is consistent with the intended application. The application of this methodology is also within the limitations of the SER. Therefore, the implementation of Addendum 1 to WCAP 12488-A does not result in a departure from a method of evaluation described in the UFSAR.*

*

*Review of Procedure LS-AA-104 Revision 3 steps 4.1.5, 4.3.1, 4.3.7, and 4.4.2 has determined that the proposed activity is encompassed by a response to 50.59 Evaluation Question 8.*

*

**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.** | |
| ✓ | **50.59 Evaluation** | **50.59 Evaluation No.** | **BRW-E-2003-220 / 6G-03-0008** | **Rev.** | **0** |

# 50.59 REVIEW COVERSHEET FORM



**Station:** Byron

**Activity/Document Number:** Technical Requirements Manual (TRM) Change 04-001 **Revision Number:** N/A

**Title:** TRM Change 04-001, 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 B2R11 from 100 hours to 63 hours:

Byron TRM Section 3.9.a, "Decay time," TLCO states "The reactor shall be subcritical for ≥ the last 100 hours (≥ 56 hours for B1R12)." This activity will revise this statement to replace "(≥ 56 hours for B1R12)" with "(≥ 63 hours for B2R11)."

Byron TRM Section 3.9.a Condition A states "Reactor subcritical for < 100 hours (< 56 hours for B1R12)." This activity will revise this statement to replace "(< 56 hours for B1R12)" with "(< 63 hours for B2R11)."

Byron TRM Surveillance TSR 3.9.a.1 states "Verify the reactor subcritical ≥ 100 hours by confirming the date and time of subcriticality. (≥ 56 hours for B1R12)". This activity will change "(≥ 56 hours for B1R12)" to "(≥ 63 hours for B2R11)".

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

It is anticipated that during B2R11, 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 B2R11 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 63 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 total number of fuel assemblies to be transferred, and the transfer rate, remains the same. There is no effect on the failure probabilities of the SFP cooling system.

# 50.59 REVIEW COVERSHEET FORM



**Station:** Byron

**Activity/Document Number:** Technical Requirements Manual (TRM) Change 04-001 **Revision Number:** N/A

**Title:** TRM Change 04-001, TRM Section 3.9.a, "Decay Time"

Revision 000G 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 maximum calculated local water temperature, the maximum calculated cladding temperature, and the actual maximum clad heat flux remained 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 63 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. Revision 000G to calculation BRW-00-0010-M/BYR2000-007 showed the maximum calculated clad temperature remains less than the mean clad temperature during power operation (700°F) and well below the design basis limit of 2200°F. The calculation also showed the actual maximum clad heat flux is more than 150 times lower than the critical heat flux required for Departure from Nucleate Boiling. 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.)

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|---|---|---|---|---|---|
| [ ] | Applicability Review | | | | |
| [ ] | 50.59 Screening | 50.59 Screening No. | | Rev. | |
| [X] | 50.59 Evaluation | 50.59 Evaluation No. | 6G-04-0001 | Rev. | 0 |