Exelon Nuclear

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U.S. Nuclear Regulatory Commission
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Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Subject: 10 CFR 50.59 Biennial Report

Pursuant to the requirements of 10 CFR 50.59, "Changes, Tests and Experiments," paragraph (d)(2), Braidwood Station is providing the required biennial report for Facility Operating License Nos. NPF-72 and NPF-77. This report is being provided for the time period of June 19, 2004 through June 18, 2006, and consists of the coversheets for changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR), and test or experiments not described in the UFSAR.

Please direct any questions regarding this submittal to Mr. Dale Ambler, Regulatory Assurance Manager, at (815) 417-2800.

Respectfully,

for

Thomas Coutu
Site Vice President
Braidwood Station

Report of Evaluations

15-Dec-06

Total Records = 14

Tracking Number / Description	Date	Closed	Unit	Preparer & Depart.	Reviewer	UFSAR Rev?	DRP Number	System Effected	EPN Number	Initiating Activity & Tracking Number	Other Supporting Documentation
BRW-E-2004-183 TRM Change #04-009 Change ICDT for A1R11 from 100 hrs to 65 hours	3/31/2004	Yes	1	PANICI, JOHN MOD DESIGN	RADICE, DANIEL	No		FC	None	Tech Reqmts Manual 04-009	
BRW-E-2004-196 CANCELLED	9/17/2004	No	1/2	STEVENS, TYRONE L BRD REACTOR ENGINEERING	SAHADEWAN, KA	No		RX1		Other	
BRW-E-2004-254 Relocate 1/2FT-RF008 bubbler tube, and replace the dP Containment Sump Narrow Level transmitters with GEMS float/reed switch transmitters	2/20/200	Yes	1/2	WUNDER, ROBERT G MOD DESIGN		Yes		PC	1/2F -RF008, 1/2L -PC002/003	Modifications EC 350243	
BRW-E-2004-272 Revise the 1C HD flash tank level setpoint and allow use of emergency level control valve as the normal means of controlling level	2/15/200	No	1	MATTHEWS, JOHN L PLANT ENGINEERING	WOLFF, NATHAN	No		HD	1HD02TC	Setpoint Change (SS EC 352955	
BRW-E-2005-9 Cancelled - item taken out in error - not needed	1/14/2005	No	2	BERGNER, JEFFREY L MOD DESIGN	YBARRA, ROQUE	No		RC	1LI-RC020	Temporary Alteration 353244	
BRW-E-2005-11 TCCP 353321	1/16/2005	Yes	2	PANICI, JOHN MOD DESIGN	GOSNELL, JAME	No		DG	2DG01KB	Temporary Alteration 353321	
BRW-E-2005-70 TRM 05-003	3/21/2005	Yes	0	PANICI, JOHN MOD DESIGN	RADICE, DANIEL	No		FC	None	Other TRM 05-003	
BRW-E-2005-81 Revision of TRM TLCO 3.4.e to add a note to permit cycling of individual head vent valves in modes 3 and 4 to reduce potential seat leakage or to perform ISI inspections.	4/5/2005	Yes	1/2	COLE, THOMAS E PLANT ENGINEERING	BELAIR, RAYMO	No		RC		Tech Reqmts Manual 3.4.e	

Tracking Number / Description	Date	Closed	Unit	Preparer & Depart.	Reviewer	UFSAR Rev?	DRP Number	System Effected	EPN Number	Initiating Activity & Tracking Number	Other Supporting Documentation
BRW-E-2005-117 this evaluation number was taken out by mistake.	5/9/2005	No	0	KOENIG, ROBERT D MOD DESIGN	YBARRA, ROQUE	No		None	n/a	Other	
BRW-E-2005-143 Support EC 355995 which performs an interim abandonment on the Acid and Caustic Regeneration Equipment in the MUDS	5/27/2005	Yes	0	MATTHEWS, JOHN L	HILDEBRANT, DO	Yes	11-030	WM	0WM04PA/B and 0WM16PA/B	Exempt Change 355995	Later determined that a Screening was only needed. Did not need to perform Evaluation.
BRW-E-2005-165 Braidwood NAC-LWT Cask Operating Procedure	8/8/2005	Yes	1	STEVENS, TYRONE L BRD REACTOR ENGINEERING	SAHADEWAN, KA	No		RX1		Procedure Revision BwOR NAC-414	
BRW-E-2006-61 Letdown Booster Pump 1CV03P Post Maintenance Test	3/10/2006	Yes	1	ZECCA, JOSEPH L		No		CV & RH	1CV03P	Other SPP-05-011	
BRW-E-2006-74 Reduced Time to start core offload during A1R12. In support of TRM change #06-009	3/19/2006	Yes	1	PANICI, JOHN MOD DESIGN	RADICE, DANIEL	No		FC	N/A	Tech Reqmts Manual 06-009	
BRW-E-2006-88 Addition of temporray tanks to store processed liquid radwaste.	4/6/2006	Yes	1/2	PANICI, JOHN MOD DESIGN	ROSCZYK, RAYM	No		WX	N/A	Temporary Alteration 358522/358725	

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Braidwood Unit 1

Activity /Document Number: TRM Change #04-009 **Revision Number:** N/A

Title: Change In-Core Decay Time for A1R11

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

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

The proposed activity makes the following Technical Requirements Manual (TRM) changes to reduce the minimum required In-Core Decay Time (ICDT) for A1R11 from 100 hours to 65 hours:

- **Braidwood TRM Section 3.9.a, "Decay time," states "The reactor shall be subcritical for ≥ the last 100 hours (≥ 65 hours for A2R10)." This activity will revise this statement by replacing "(≥ 65 hours for A2R10)" with "(≥ 65 hours for A1R11)."**
- **Condition A under TRM 3.9.a states "Reactor subcritical for < 100 hours (< 65 hours for A2R10)". This activity will replace "(< 65 hours for A2R10)" with "(< 65 hours for A1R11)".**
- **Surveillance requirement TSR 3.9.a.1 will be revised by replacing "≥ 65 hours for A2R10" with "≥ 65 hours for A1R11".**

Reason for Activity:

Discuss why the proposed activity is being performed)

It is anticipated that during A1R11, 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 TRM fuel movement ICDT constraint of 100 hours after reactor shutdown.

The Byron and Braidwood spent fuel pool cooling design basis analysis is based on the minimum ICDT of 100 hours prior to starting fuel transfer, however an outage specific evaluation has been performed to support a reduced ICDT for A1R11.

The current radiological design basis analysis for the Fuel Handling Accident is based on a minimum decay time of 48 hours prior to movement of irradiated fuel assemblies within the reactor vessel. As part of the Power Uprate Project, the radiological consequences of a Fuel Handling accident were evaluated and it was demonstrated that an ICDT of greater than or equal to 48 hours is acceptable for radiological considerations (Reference 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). Sections B3.9.4 and B3.9.7 of the Braidwood Technical Specification Bases are not being revised since the minimum ICDT for radiological considerations is not being revised and the revised ICDT for A1R11 still meets this constraint.

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

`e proposed changes will allow starting A1R11 Fuel offloading activities earlier than 100 hours.

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Activity /Document Number: TRM Change #04-009 **Revision Number:** N/A

Title: Change In-Core Decay Time for A1R11

Occupational Radiation Dose

Beginning core alteration and fuel transfer operation as early as 65 hrs after shutdown is not expected to increase the occupational dose. Per UFSAR tables 12.3-1 and 12.3-2, areas in the plant are divided 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 area affected by the defueling operation is designated as High Radiation area (Zone III). Access to these areas is controlled in accordance with station procedures and RWP. 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.

Normal Plant operation is not changed. Core defueling activities continue to follow approved station procedures. The maximum fuel transfer rate is administratively controlled to eight assemblies per hour. The only difference is that the start of the defueling activities may be as early as 65 hours after achieving subcriticality for A1R11.

Summary of Conclusion for the Activities 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.)

ıis activity does not increase the frequency of occurrence of a Fuel Handling Accident or a Loss of ɔpent Fuel Pool Cooling event, or increase the likelihood of occurrence of a malfunction of an SSC important to safety. This is based on the outage specific evaluation that concludes the total heat load in the Spent Fuel Pool as a result of the reduced ICDT is bounded by the total heat load specified in the design basis analysis and is based on the fact that all refueling activities will continue to use the normal refueling procedures and equipment.

The Fuel Handling Bldg radiation monitors and ventilation system are not adversely impacted. The monitors are not degraded by the radiation field expected due to the shorter ICDT. The likelihood of a malfunction of the Spent Fuel Pool Cooling system is not increased since the heat load on the system due to the reduced ICDT is bounded by the design basis analysis. Since the maximum bulk water temperature is not affected, the qualification of the spent fuel pool structure is not degraded.

This activity does not result in an increase in the consequences of an accident or in the consequences of a malfunction of an SSC important to safety. The offsite dose resulting from a Fuel Handling Accident considering a minimum In-Core Decay Time of 65 hours is bounded by the design basis Fuel Handling Accident dose with a minimum ICDT of 48 hours.

This activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR as there is no new equipment being introduced, and all existing fuel transfer equipment is being operated using existing procedures.

This activity does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR. The increase in heat load in the Spent Fuel Pool has been evaluated; although some input parameters have been changed, the resulting impact on the SFP bulk water temperature analysis is bounded by the design basis analysis. In addition, the local water temperature, fuel cladding nperature, and maximum heat flux have also been evaluated and have been found to be acceptable.

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Title: Change In-Core Decay Time for A1R11

The adequacy of the reduced ICDT for A1R11 is based on the additional margin remaining in background decay heat since the SFP is not filled to its capacity.

The reduction in ICDT does not result in a change in the internal containment pressure that would represent a challenge to the containment design basis limit of 50 psig. The maximum cladding temperature for the spent fuel is well below the design basis limit of 2,200 °F. Therefore, the reduced ICDT does not result in exceeding design basis limits for a fission product barrier. In addition, this activity does not make any physical changes to the spent fuel, the containment or the RCS boundary that would result in altering their design basis limit.

This activity does not change the method of evaluation for the Spent Fuel Pool Cooling System described in the UFSAR or in the SER for the Power Uprate Project. 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 design basis analysis.

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.**	BRW-E-2004-183	**Rev.**	0

BRW-E-2004-254

50.59 REVIEW COVERSHEET FORM

Station: Braidwood Units 1 and 2

Activity/Document Number: Unit 1 EC 350253 and Unit 2 EC 354786 **Revision Number:** 0

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
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Background Information

The scope of this Engineering Design Change includes: 1) relocating the Containment Floor Drain Sump Leak Detection bubbler tube for 1(2)F -RF008 from inside the RF008 weir cavity to the main sump cavity. This includes replacing the existing transmitter 1(2)F -RF008, with a transmitter having a larger pressure range, 2) replacing the Containment Floor Drain Sump Narrow Range dP Level Transmitters 1(2)LT-PC002 and 1(2)LT-PC003, with Gems magnetic float type level transmitters and converter. This includes adding a new signal converter and changes internal to cabinets 1(2)PA33J and 1(2)PA34J , 3) replacing the Penny & Giles MCR recorder 1(2)FR-RF008, with a Yokogawa model recorder. This includes recorder programming changes, 4) modifying the Containment Floor level input and actuation point for the ECCS pump suction indicating lights (5th light) 1(2)LL-SI075A and 1(2)LL-SI075B, and 5) replacing 2 conductor cable with 3 conductor cable for 1(2)LT-PC006 and 1(2)LT-PC007, as a contingency for future replacement. Activities 4 and 5 will be evaluated in 50.59 screening BRW-S-2005-59. Only activities 1, 2, and 3 will be evaluated by this Safety Evaluation.

Physical and Functional Description of the Existing Containment Floor Drain Sump Leak Detection Instrumentation Loop(s) 1(2)F -RF008:

The existing Containment Floor Drain Sump Leak Detection Instrument loop 1(2)F -RF008, consist of a bubbler tube positioned in the RF008 sump to measure unidentified leakage collected in the sump cavity. As the unidentified leakage level rises and falls, a crest is created across the weir slot that is measured as a change in backpressure by the bubbler tube system. The pressure as a function of level rise across the weir slot, is sensed by the transmitter that sends a corresponding signal to a function generator in cabinet 1(2)PA20J, which converts the signal to flow rate. The flow signal is transmitted to a recorder located on panel 1(2)PM12J and also to a high flow alarm annunciator on panel 1(2)PM06J.

The 1(2)F -RF008 Instrument Loop functions solely to satisfy Technical Specification 3.4.15, "RCS Leakage Detection Instrumentation," which requires the containment sump monitor to be operable or enter a 30 day LCO. 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 within 1 hour or less. To meet Regulatory Guide 1.45, the 1(2)F -RF008 loop provides a High Flow rate Alarm (set for 1 gpm), and flow rate indication on a Digital Chart Recorder. 1(2)F -RF008 determines flow rate 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. There is no automatic trip function or control function performed by this instrument loop. The RCS leakage detection requirement is satisfied by either the High Flow rate Alarm or through hourly operator monitoring of flow rate indications. Tech Spec 3.4.15 requires only one containment sump monitor operable [1(2)F -RF008, 1(2)L -PC002, or 1(2)L -PC003] and one containment atmosphere radioactivity monitor operable. UFSAR Appendix A, page A1.45-2 (Regulatory Guide 1.45 Clarification), states that the 1(2)F -RF008 Loop *"is designed to remain functional after a Safe Shutdown Earthquake (SSE) and is powered by non-ESF buses."* Instrument loop 1(2)F -RF008 and its associated power supply are not required to satisfy the design criteria of UFSAR Section 3.2.1.1 for Safety Category I functions, and therefore is not

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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

classified as safety related (reference Byron CR 100121 and Operability Determination 02-005). Instrument loop 1(2)F -RF008 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 alarm and indication function. In the event that this instrument loop fails, Tech Spec 3.4.15 Bases permits use of either 1(2)L -PC002 or 1(2)L -PC003 for detecting unidentified leakage of 1 gpm within one hour based on the relationship of floor drain sump level change over time.

Physical and Functional Description of the Containment Floor Drain Sump Level Instrumentation Loops 1(2)L -PC002 and 1(2)L -PC003:

The existing Containment Floor Drain Sump Level Instrument Loops consist of Barton model 764 dP transmitters with a remote capillary tube and bellows system. The remote bellows is located at the bottom of the sump and is used to measure hydrostatic pressure as a function of level. The transmitters measure the differential pressure (low side at transmitter) and sends a signal to cabinets 1(2)PA33J and 1(2)PA34J respectively. From here the signal is converted to a usable form and indicated as level at panel 1(2)PM06J, and at the Plant Process Computer, SPDS, and/or corresponding flow rate at the plant graphic display system (PIOILS).

The 1(2)L -PC002/003 Loops (Redundant Containment Floor Drain Narrow 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. The Containment Floor Drain sump Narrow Range Level Loop is used to provide post accident monitoring of containment water level and fulfill the requirements of NUREG-0737 and Regulatory Guide 1.97.

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, the instrument loops as originally designed, cannot directly calculate and display flow rate (except at PIOILS, the plant graphic display system), and do not contain a High Flow rate Alarm function capable of being set at 1 gpm in the same manner as the 1(2)F -RF008 Loop. Since 1(2)L -PC002 and 1(2)L -PC003 currently do not have alarm capability, operators must monitor 1(2)L -PC002 or 1(2)L -PC003 at least every hour to assure meeting the function of detecting a 1 gpm RCS leak within one hour when instrument loop 1(2)F –RF008 is not operable.

Description of Activity:

Design Change Description for Instrument Loop 1(2)F -RF008:

The Engineering Design Change will disable/abandon the existing "weir plate" method of determining RCS leakage. The 1(2)F -RF008 RCS Leakage Detection function (flow rate indication and high flow rate alarm) will be modified by relocating and extending the RCS Leakage Detection bubbler tube into the bottom of the Containment Floor Drain Sump. Tubing will be routed to the main sump using the existing tubing track used for the remote seal capillary lines associated with dP transmitter 1(2)LT-PC003. Sump flow rate indication and high flow alarm will become a function of sump level change over time, versus the previous method of flow rate measurement that was differential pressure across the weir slot. To accomplish this change, the existing Containment Floor Drain leak detection loop transmitter [1(2)F -RF008] will be replaced with one of the same manufacturer and type having a wider pressure range to accommodate the larger level span of the containment floor drain sump. The new bubbler tube/transmitter and recorder configuration will satisfy the requirements of

Station: Braidwood Units 1 and 2

Activity/Document Number: Unit 1 EC 350253 and Unit 2 EC 354786 **Revision Number:** 0

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

Technical Specification 3.4.15, "RCS Leakage Detection Instrumentation," by measuring unidentified leakage within the licensing requirements of 1 gpm within one hour.

Design Change Description for Instrument Loops 1(2)L -PC002 and 1(2)L -PC003:

The existing Containment Floor Drain Sump Differential Pressure type level transmitters 1(2)LT-PC002 and 1(2)LT-PC003, will be removed and replaced with two new GEMS magnetic float type resistance level transmitters and associated signal conditioners. The new GEMS transmitters will be mounted to a newly installed vertical tube steel supports attached to the containment sump liner. The new GEMS level system will continue to satisfy the requirements of Technical Requirements Manual (TRM) 3.3.i, "Post Accident Monitoring (PAM) Instrumentation."

Tech Spec 3.4.15 Bases permits the use of the 1(2)L -PC002 or 1(2)L PC003 for detecting unidentified leakage of 1 gpm within one hour as a measure of floor drain sump level change over time. The existing instrument loops 1(2)L -PC002 and 1(2)L -PC003, are approved for use to satisfy RCS Leakage detection upon failure or loss of the 1(2)F -RF008 instrument function. Direct level measurement will continue to be available for operators to assess changes. An added feature will be the ability of the operators to manually switch the recorder function from a "Normal mode," which calculates flow rate from the primary leak detection loop 1(2)F –RF008, to the "GEMS mode," which calculates flow rate as a function of level input from sump level instrument loops 1(2)L –PC002/3.

Design Change Description for Recorder 1(2)FR-RF008:

The Penny & Giles Digital Chart Recorder 1(2)FR-RF008, will be replaced with a Yokogawa Digital Chart Recorder that is capable of performing the mathematical functions needed to convert level readings into flow rate and provide the corresponding indication and alarm actuations for instrument loops 1(2)F –RF008 and 1(2)L-PC002/3.

Alarm outputs from the Yokogawa recorder will be wired into the existing MCR Annunciator box to provide both a high flow rate 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 that could impact sump flow rate calculations and response time as described in this evaluation). The new recorder will be programmed to prevent faulty flow rate 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 flow rate indication. After the sump pump down "transient" is complete, it may take a period of time for steady state flow rate indication to return to its pre-pump down value. The recorder will be programmed to lock out the high flow alarm actuation until steady state flow rate indication can return after sump pump down. The alarm lockout times will be determined from modification testing, and will be selected to not exceed the RCS Leakage Detection response time requirements of 1 hour or less, when combined with other parameters that would delay the response time. The Yokogawa recorder will be programmed to directly calculate flow rate and provide a high flow rate alarm based on the elapsed time between level steps of the float passing the reed switches spaced at ½" intervals within the GEMS [1(2)LE -PC002/003] transmitters. This function will normally be disabled, but is available as a backup if the normal leak detection flow loop 1(2)F -RF008 is inoperable.

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Station: Braidwood Units 1 and 2

Activity/Document Number: Unit 1 EC 350253 and Unit 2 EC 354786 Revision Number: 0

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

Reason for Activity:

Containment Floor Drain Sump Leak Detection Instrumentation Loop(s) 1(2)F -RF008:

The existing 1(2)F -RF008 loop has been declared inoperable numerous times as a result of weir plate blockage, especially when non-RCS leakage continuously flows through the weir box. This brings in the high flow rate alarm, and the instrument loop is declared inoperable. This requires that Operators check for RCS leakage every hour using alternative instrumentation [1(2)L -PC002 or 1(2)L -PC003] to ensure compliance with Technical Specification LCO 3.4.15. In addition, the 7300 NCH card used for converting measured dP to flow, has a history of frequent failure. The modified leak detection instrument loop will not be prone to weir plate blockage since the bubbler tube will be relocated outside the weir box, and the NCH card will be removed and the conversion function replaced by the math capability of the recorder software.

Containment Floor Drain Sump Level Instrumentation Loops 1(2)L -PC002 and 1(2)L -PC003:

The Barton model 764 dP transmitters [1(2)LT-PC002/3] that are currently used to measure containment floor drain sump level, requires IM technicians to take large dose when they enter the sump to calibrate the loop. Following the initial setup, the GEMS transmitters do not require calibration at the transmitters. Because of their stability and the principle of operation, the technicians do not need to enter the sump every outage. The GEMS floats will occasionally be checked functionally to measure any potential resistor degradation, but this will be less frequent and less time consuming then the Barton transmitters, therefore minimizing the dose that the IM's receive.

Effect of Activity:

Regulatory Guide 1.45 states,

> *"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 described in EC 350253 and EC 354786 are consistent with the requirements of Reg. Guide 1.45, UFSAR Appendix A, and Technical Specification Bases 3.4.15. There will be a known relationship between sump water level rise and unidentified leakage flow rate. This known relationship will replace the 1(2)RF008 weir box flow rate as measured by dP across the slot. The modified 1(2)FT-RF008 bubbler tube/transmitter and new recorder are consistent with the requirements described in Regulatory Guide 1.45. The revised instrument scaling relationship will readily allow converting level indication into the desired common denominator (flow rate). The response time of the new method of leakage detection may be increased when compared to the weir method of flow measurement because of the sump capacity compared to the weir box capacity, but the overall requirements of detecting 1 gpm within 1 hour will continue to be in compliance. This has been confirmed by engineering analysis and will also be verified by modification testing.

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Field testing will verify that the alarm and recorder indication accurately display the criteria flow of 1 gpm within one hour prior to declaring the instrumentation operable. Affects of sump pump down on flow indication and response time will be tested and verified. Operating procedures will be revised to assure the new bubbler tube/transmitter and recorder configuration can be utilized in a manner essentially the same as current practice.

The proposed activity will require a UFSAR change to reflect the new configuration for description of leak detection instruments as the 1(2)F -RF008 weir box design is described in detail. Additionally, description of Post Accident Monitoring requirements for containment water level will require a UFSAR change to reflect the new instrumentation configuration (float-type level transmitters with signal conditioners versus dP remote bellows type transmitters). However, the new 1(2)F -RF008 bubbler tube/transmitter and recorder configuration will perform the same function as the existing RCS leak detection loop, and the new 1(2)L -PC002 and 1(2)L - PC003 level instrumentation will perform the same function as the existing Post Accident Monitoring loops. Therefore, there will be no functional requirement changes as a result of this modification.

No changes are required to the description or function of leakage detection instrumentation requirements of Technical Specification 3.4.15. Technical Specification Basis will require a change to take credit for the leak detection flow alarm function of 1(2)L -PC002/003. No changes are required to the Post Accident Monitoring Functions of TRM 3.3.i.

The existing containment floor drain sump flow channel and its power supply are not required to comply with 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 classified as safety related. This statement applies to the existing high flow rate alarm and will apply to the modified high flow rate 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 that would alert operators to the loss of the associated non-ESF power supply to the recorder.

Summary of Conclusion for the Activity's 50.59 Review:

The proposed design change is being implemented to improve instrument reliability. Braidwood and Byron have documented numerous problems with both the leak detection weir method of measurement, and the containment floor drain sump level instrumentation.

1(2)F –RF008 functions to detect Reactor Coolant Pressure Boundary (RCPB) leakage as soon after it starts as practical to preclude the potential for a larger RCPB failure (i.e., Leak Before Break analysis prior to analyzed Loss of Coolant Accident) in compliance with Regulatory Guide 1.45, revision 0, "*Reactor Coolant Pressure Boundary Leak Detection Systems*". Braidwood Technical Specification Bases B.3.4.15 states,

> *"Leak detection systems must have the capability to detect significant RCPB degradation as soon after occurrences as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified leakage".*

50.59 REVIEW COVERSHEET FORM

Station: **Braidwood Units 1 and 2**

Activity/Document Number: **Unit 1 EC 350253 and Unit 2 EC 354786** **Revision Number:** **0**

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

and also,

> "*.... safety analysis does not address operational leakage. However, other operational leakage* (meaning other than primary to secondary) *is related to the safety analysis for LOCA".*

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. Several diverse methods are used at Braidwood. One acceptable method of which is to passively indicate unidentified leakage flow rate as a function of sump level change and alarm when the flow rate reaches 1 gpm within one hour as specified in section 5.2.5.1.d of the B/B UFSAR, which complies with Regulatory Guide 1.45.

Compliance with Regulatory Guide 1.45 to detect a 1 gpm RCS leak within 1 hour, will be confirmed by modification testing following installation. A known flow rate will be used to fill the sump and measurements taken to determine the expected flow rate, and the alarm response time. Modification testing will be performed on both the 1(2)F –RF008 instrument loop, and the 1(2)L –PC002/3 instrument loops.

The "Leak-Before-Break" analysis relies on diversity as well as the capability of complying with the R.G. 1.45 criteria of detecting a 1 gpm RCS unidentified leak within one-hour. Leak detection diversity remains unchanged and the modified sump flow rate instrument loop will be capable of responding to 1 gpm within one hour.

Failure modes for both the leak detection instrumentation and the PAM instrumentation have been evaluated in Design Consideration Summary of the applicable EC's. It was concluded that there would not be an increase in malfunctions with the newly installed instrumentation.It was also concluded that because of the passive nature of these loops, which do not physically or functionally interact with other SSCs that could initiate or impeded mitigating an accident, there will be no impact on other SSCs important to safety.

This design change will not require a Technical Specification or Operating License change. The Braidwood UFSAR and Technical Specification Bases will be revised to describe the addition of the leak detection flow alarm function of 1(2)L -PC002/003 when used in the leak detection mode.

In conclusion, the proposed activity does not: result in more than a minimal increase in the frequency or the consequences of occurrence of an accident previously evaluated in the UFSAR; create a possibility for an accident of a different type than any previously evaluated in the UFSAR; result in more than a minimal increase in the likelihood of occurrence or the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR; create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR; does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered; or result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses. Therefore, the proposed modification may be implemented without prior NRC approval.

50.59 REVIEW COVERSHEET FORM

Station: **Braidwood Units 1 and 2**

Activity/Document Number: **Unit 1 EC 350253 and Unit 2 EC 354786** **Revision Number:** **0**

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

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.**	**BRW-E-2004-254**	**Rev.**	**0**

BRW-E-2004-272

50.59 REVIEW COVERSHEET FORM

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Station/Unit(s): Braidwood / Unit 1

Activity/Document Number: SSCR Type EC 352955 **Revision Number:** 000

Title: Revise Setpoint For 1C Flash Tank Emergency Level Controller and Low Level Alarm

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.)

Braidwood Station has three strings of low-pressure heaters (A, B, and C) each with four heaters (1 thru 4), a flash tank and a drain cooler. Condensate Booster (CB) system flow is through the tube side with extraction steam and condensate flow from the upstream heater on the shell side. The number 2 heater drains to the flash tank. Within the flash tank, a portion of the condensate flashes to steam flows to the number 1 heater. The condensate from the number 1 heater returns to the flash tank and the flash tank then drains to the drain cooler. The drain cooler drains to the condenser. There is no extraction steam supply for the drain cooler. An emergency drain valve allows condensate from the flash tank to drain directly to the condenser (bypassing the drain cooler) on a high-2 level condition in the flash tank. The input from the number 2 heater to the flash tank is also isolated under a high-2 condition.

The level control loop for each HD Flash Tank consists of a level sensing device (LE), a level transmitter (LT), a signal converter (LY) and a normal level controller (LC) and an emergency level controller (LC).

Engineering Change (EC) 352955 is a Scaling and Setpoint Change Request (SSCR) that will result in the following changes:

- Revise the 1C Heater Drain (HD) Flash Tank emergency level controller (1LC-HD283B) setpoint from 90% to 50%
- Revise the 1C HD Flash Tank low level alarm (1LSL-HD283) from 68% to 40%
- Isolate the normal drain valve 1HD029C from the 11C drain cooler to the condenser thereby requiring the 1C HD Flash Tank to drain directly to the condenser through the emergency level control valve 1HD094C

These changes are being evaluated for a period of up to a full operating cycle of approximately 18-months (however, repairs are expected to be completed prior to that time and the system returned to it's original condition). This is based on the satisfactory completion of certain additional actions (e.g., inspections, additional evaluations) as described in the Safety Evaluation. This Safety Evaluation is applicable to any Operating Procedure revisions required to implement EC 352955.

With these changes in place, the level in the 1C HD Flash Tank will be maintained at 50% of the level tree span with flow through the shell side of the 11C drain cooler isolated. The 1C HD Flash Tank is a vertical tank with a 96-inch diameter shell, hemispherical heads and an overall height of approximately 14-feet. The level tree for the tank has a span of approximately 53 inches with the 0% corresponding to a level of 8-inches above the lower head to shell connection and 100% corresponding to a level 5'-1" above this same head to shell connection. The current settings are: 68% low level alarm, 80% normal level controller setpoint, 90% high level alarm and emergency level controller setpoint, and 95% high-2 actuation which isolates the input from the 12C heater and opens the flash tank emergency drain valve to the condenser. The actual level change in the tank will be approximately 16-inches lower.

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

The level control system for the 1C HD Flash Tank is exhibiting unreliable performance due to a degraded control loop. The condition is limiting the normal and emergency level controller (1LC-HD283A/B) to less than the full span of the level loop. This degraded condition was a contributing factor for a high-2 actuation on the 1C HD Flash Tank and the 12C, 13C and 14C Low Pressure (LP) Heaters on 12/13/2004 (Reference IR 281958, 1C HD FLASH TANK HI-2 CAUSES LOSS OF FEEDWATER HEATERS). Repair of the degraded control loop on line is considered a high-risk activity because the instruments are in the same cabinet as the 1B HD Flash Tank level control instruments.

50.59 REVIEW COVERSHEET FORM

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Station/Unit(s): Braidwood / Unit 1

Activity/Document Number: SSCR Type EC 352955 **Revision Number:** 000

Title: Revise Setpoint For 1C Flash Tank Emergency Level Controller and Low Level Alarm

The changes being affected by SSCR EC 352955 will improve operation of the 1C HD Flash Tank level control loop until the degraded condition can be repaired. The benefits are:

- With a lower level control setpoint, the existing control loop will allow the emergency level control valve to operate over its entire span and achieve the full open position when and if required by system conditions
- By isolating the 11C drain cooler, the normal operating level (normal level controller setpoint) can be lower than with the cooler in service. With the cooler isolated, there is no minimum required level in the flash tank to protect the cooler.

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

This activity will have the following affect on the plant.

- Loss in plant efficiency equivalent to 0.3 MWe from isolating the 11C drain cooler.
- Increased load on the 11C heater
- Increased wear on the 1C HD Flash Tank Emergency Level Control valve 1HD094C and the associated piping
- Increased heat load on the condenser
- Increased loading on the1C HD Flash Tank emergency level control valve nozzle connection to the main condenser
- 11C Drain Cooler will be partially filled (75%) with flow isolated to the shell side.

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.)

A 50.59 Evaluation concluded that this change could be made pursuant to 10CFR50.59 because it did not require a change to the Technical Specifications nor did it meet any of the eight criteria listed in Section (c)(2). The change will have a minor impact on several sections of the UFSAR as follows:

- UFSAR Figure 10.1-1 shows the 1C HD Flash Tank draining to the 11C drain cooler. This is being impacted because the flash tank will drain directly to the condenser and bypass the cooler. The evaluation performed under EC 352955 concluded that this would not adversely affect plant operation if limited to one operating cycle (18-months).
- UFSAR Figure 10.1-2a for the Unit 1 Heat Balance. This is being impacted because the temperature enthalpy of the condensate draining from the number 1 drain cooler/heater will be higher than that shown. It will be increased by the amount of energy normally removed in the drain cooler. This will not adversely affect plant operation. Plant output will be minimally impacted by a reduction of 0.3 MWe.
- UFSAR Section 10.4.7.6 states that a high level on a heater shell side will cause the condensate to dump to the condenser. For the 11C heater and 1C HD Flash Tank, the contents are already being dumped to the condenser versus the 11C drain cooler. The evaluation performed under EC 352955 concluded that this would not adversely affect plant operation if limited to one operating cycle (18-months).

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Braidwood / Unit 1

Activity/Document Number: SSCR Type EC 352955 **Revision Number:** 000

Title: Revise Setpoint For 1C Flash Tank Emergency Level Controller and Low Level Alarm

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.**	**BRW-E-2004-272**	**Rev.**	**000**

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Braidwood U-2

Activity /Document Number: TCCP #353321 and associated procedure revisions **Revision Number:** 0

Title: **Install blind flanges to provide additional isolation for the suction and discharge piping from the 2B diesel generator motor driven jacket water pump and implement compensatory actions to maintain the Operability of the 2B EDG**

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

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

This activity includes the following:

1. TCCP 353321
   With the jacket water circulating pump removed, install blind flanges on the suction side and discharge side of the 2B Emergency Diesel Generator (EDG) motor-driven jacket water pump.
2. BwOP DG-1, DIESEL GENERATOR ALIGNMENT TO STANDBY CONDITION, Revision 23a
3. BwOP DG-3, FILLING THE DIESEL GENERATOR JACKET WATER SYSTEM, Revision 8a
4. BwOP DG-11, DIESEL GENERATOR STARTUP, Revision 29a
5. BwOP DG-12, DIESEL GENERATOR SHUTDOWN, Revision 22a

Procedure BwOP DG-1 is being revised to add a note to reflect that (1) the 2B EDG will not have auto make-up capability to maintain standpipe level while EC 353321 is installed, (2) the applicable steps of BwOP DG-3 will be used to fill the standpipe manually if required and (3) the steps requiring verification that the jacket water circulating pump control switch and the jacket water heater control switch are in "Auto" are not applicable to the 2B EDG.

Procedures BwOP DG-1, BwOP DG-3, BwOP DG-11, and BwOP DG-12 are being revised to add a note to reflect that (1) the 2B EDG will not have auto make-up capability to maintain standpipe level while EC 353321 is installed, and (2) the applicable steps of BwOP DG-3 will be used to fill the standpipe manually if required. Other temperature considerations (Delta-Temperature between jacket water and lube oil) identified in the Adverse Condition Monitoring Plan are also added for informational purposes only in the procedures as applicable.

A specific step was added to BwOP DG-3 to provide instructions to manually fill the jacket water system for the 2B EDG while EC 353321 is installed.

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

The motor for the jacket water circulating pump has been removed and has been shipped for repairs at an offsite facility. The flanges and procedure changes are needed, on a temporary basis, to implement actions intended to restore the 2B EDG to operability until the repaired pump is installed.
The function of the jacket water pump and its associated heater will be temporarily replaced by periodically running the 2B EDG. This action will maintain the temperature of the jacket water above 105 °F, as outlined in the Adverse Condition Monitoring Plan developed in accordance with the requirements of procedure OP-AA-108-111. This temperature is above the lube oil and jacket water minimum temperature of 100 °F that is acceptable for operability of the EDG (Reference: Contact Report - Glenn Miller of Cooper Energy dated 11/30/88 at 0930).
The function of the automatic standpipe level control is being replaced by manual operator action, initiated by an installed level alarm, and controlled by an approved procedure.

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Rev.

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

The temporary change and procedure revisions will result in a short-term increase in operator involvement in maintaining the 2B Emergency Diesel Generator in a stand-by condition. Operators will have to periodically run the diesel engine in accordance with approved procedures to maintain the diesel in a standby (warm) condition thus helping to ensure a quick start should the engine receive an auto-start signal. Since the engine is housed in a temperature-controlled room, it is expected that the engine will have to be run no more than once each day. The loss of automatic make-up to the jacket water stand pipe is not expected to increase operator burden since the jacket water is not consumed by running the engine.

The expected duration of the change is less than 1 week.

Note: The Technical Specifications Bases for SR 3.8.1.2 and SR 3.8.1.7 state that the Standby condition and Normal Standby conditions mean that the diesel engine coolant and the lube oil are being continuously circulated and temperature is being maintained consistent with the manufacturer's recommended and prescribed ranges. A review of electrical schematic 4030 DG25 shows that the jacket water pump and jacket water heater cycle on/off with jacket water temperature. Therefore, the diesel engine coolant is not being continuously circulated. Issue Report #291277 has been initiated to address this condition. This discrepancy does not have any impact on the results of this 50.59 evaluation. The function of maintaining temperature by the intermittent cycling of the electric jacket water pump and its associated electric heater is acceptable based on past performance and will be fully addressed within the Issue Report resolution process.

Summary of Conclusion for the Activities 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 function of the electric jacket water pump and its associated electric heater is to maintain the diesel engine in a normal standby condition by circulating warm water through the engine cooling system when the engine is not running, to enhance quick starting capability. The pump and electric heater are designed to maintain the temperature of the diesel engine jacket water above 110 °F. Once the engine is started, the cooling water for the diesel is circulated by the engine driven jacket water pump and the electric jacket water pump serves no purpose other than to maintain a pressure boundary. The function of maintaining the engine warm is being replaced by periodically running the diesel engine. The clearance order boundary has been established at valve 2DG5046B, Butterfly valve at the suction side of the pump, and at valve 2DG5047B, check valve at the downstream side of the pump. These two valves maintain the ASME pressure boundary function with the added blind flanges as additional leak prevention features. The function of the automatic jacket water standpipe level control is replaced by an existing low level alarm and approved procedure for adding water to the standpipe by manual action. As a result of maintaining these functions, no prior NRC approval of the changes is needed.

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.**	**BRW-E-2005-11**	**Rev.**	**0**

BRW-06-2005-70

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Braidwood Unit 2

Activity /Document Number: TRM Change #05-003 **Revision Number:** N/A

Title: Change In-Core Decay Time for A2R11

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

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

The proposed activity makes the following Technical Requirements Manual (TRM) changes to reduce the minimum required In-Core Decay Time (ICDT) for A2R11 from 100 hours to 76 hours:

- **Braidwood TRM Section 3.9.a, "Decay time," states "The reactor shall be subcritical for ≥ the last 100 hours (≥ 65 hours for A1R11)." This activity will revise this statement by replacing "(≥ 65 hours for A1R11)" with "(≥ 76 hours for A2R11)."**
- **Condition A under TRM 3.9.a states "Reactor subcritical for < 100 hours (< 65 hours for A1R11)". This activity will replace "(< 65 hours for A1R11)" with "(< 76 hours for A2R11)".**
- **Surveillance requirement TSR 3.9.a.1 will be revised by replacing "≥ 65 hours for A1R11" with "≥ 76 hours for A2R11".**

Reason for Activity:

Discuss why the proposed activity is being performed)

It is anticipated that during A2R11, 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 TRM fuel movement ICDT constraint of 100 hours after reactor shutdown.

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

The proposed changes will allow starting A2R11 Fuel offloading activities earlier than 100 hours.

The Byron and Braidwood spent fuel pool cooling design basis analysis is based on the minimum ICDT of 100 hours prior to starting fuel transfer, however an outage specific evaluation has been performed to support a reduced ICDT for A2R11. Starting core offload at 76 hours after shutdown will not result in increasing the design basis heat load for the Spent Fuel Pool (SFP) Cooling System.

Moving fuel early does result in an increase in the heat load input to the Spent Fuel pool compared to starting core offload at 100 hrs or later. However, the overall heat load to the spent fuel pool will not be greater than the design basis heat load, since the actual heat load in the SFP due to previously stored fuel assemblies is less than the heat load that was included in the SFP design basis analysis. The impact on the actual spent fuel pool temperature is expected to be minimal since two (2) trains of cooling can be operated during core offloads. Limitations and alarms

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Activity /Document Number: TRM Change #05-003 **Revision Number:** N/A

Title: Change In-Core Decay Time for A2R11

related to Spent Fuel temperatures are in place during this activity. The limitation from EC #354006 for moving fuel at a rate not greater than seven (7) assemblies per hour is not expected to have any impact on refueling operations, since during the last several refueling outages this offload rate has not been exceeded.

The current radiological design basis analysis for the Fuel Handling Accident is based on a minimum decay time of 48 hours prior to movement of irradiated fuel assemblies. As part of the Power Uprate Project, the radiological consequences of a Fuel Handling accident were evaluated and it was demonstrated that an ICDT of greater than or equal to 48 hours is acceptable for radiological considerations (Reference 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). Sections B3.9.4 and B3.9.7 of the Braidwood Technical Specification Bases are not being revised since the minimum ICDT for radiological considerations is not being revised and the revised ICDT for A2R11 still meets this constraint.

<u>Occupational Radiation Dose</u>
Occupational radiation dose will remain within limits. Per UFSAR tables 12.3-1 and 12.3-2, areas in the plant are divided 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 area affected by the defueling operation is designated as High Radiation area (Zone III). Access to these areas is controlled in accordance with station procedures and RWP. 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.

Summary of Conclusion for the Activities 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 does not increase the frequency of occurrence of a Fuel Handling Accident or a Loss of Spent Fuel Pool Cooling event, or increase the likelihood of occurrence of a malfunction of an SSC important to safety. This is based on the outage specific evaluation that concludes the total actual heat load in the Spent Fuel Pool as a result of the reduced ICDT is bounded by the total heat load specified in the design basis analysis.

The Fuel Handling Bldg radiation monitors and ventilation system are not adversely impacted. The monitors are not degraded by the radiation field expected due to the shorter ICDT. The likelihood of a malfunction of the Spent Fuel Pool Cooling system is not increased since the heat load on the system due to the reduced ICDT is bounded by the design basis analysis. Since the maximum bulk water temperature is not affected, the qualification of the spent fuel pool structure is not degraded.

This activity does not result in an increase in the consequences of an accident or in the consequences of a malfunction of an SSC important to safety. The offsite dose resulting from a Fuel Handling Accident with a minimum In-Core Decay Time of 76 hours is bounded by the design basis Fuel Handling Accident dose with a minimum ICDT of 48 hours.

50.59 REVIEW COVERSHEET FORM

Activity /Document Number: TRM Change #05-003 **Revision Number:** N/A

Title: Change In-Core Decay Time for A2R11

This activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR as there is no new equipment being introduced, and all existing fuel transfer equipment is being operated using existing procedures.

This activity does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR. The increase in heat load in the Spent Fuel Pool has been evaluated; although input parameters (offload start time and fuel transfer rate) have been changed, the resulting impact on the SFP bulk water temperature analysis is bounded by the design basis analysis. In addition, the local water temperature, fuel cladding temperature, and maximum heat flux have also been evaluated and have been found to be acceptable.

The adequacy of the reduced ICDT for A2R11 is based on the additional margin remaining in background decay heat since the SFP is not filled to its capacity.

The reduction in ICDT does not result in a change in the internal containment pressure that would represent a challenge to the containment design basis limit of 50 psig. The maximum cladding temperature for the spent fuel is well below the design basis limit of 2,200 °F. Therefore, the reduced ICDT does not result in exceeding design basis limits for a fission product barrier. In addition, this activity does not make any physical changes to the spent fuel, the containment or the RCS boundary that would result in altering their design basis limit.

his activity does not change the method of evaluation for the Spent Fuel Pool Cooling System described in the UFSAR or in the SER for the Power Uprate Project. 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 design basis analysis.

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.**	BRW-E-2005-70	**Rev.**	0

BRW-E-2005-81

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: TRM TLCO 3.4.e **Revision Number: : TRM Change Request 05-004**

Title: Reactor Vessel Head Vents

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:

Braidwood Unit 1 and Unit 2 Technical Requirements Manual (TRM) Limiting Condition for Operation (TLCO) 3.4.e, Reactor Vessel Head Vents, is being revised to incorporate a TLCO note to permit vent valve cycling in Modes 3 and 4. Currently the TRM TLCO reads:

Two reactor vessel head vent paths, each consisting of two valves in series powered from emergency buses, shall be OPERABLE and closed.

Applicability: MODES 1, 2, 3, and 4.

The new TRM TLCO note will modify the TRM TLCO as follows:

The reactor vessel head vent valves may be cycled for the purposes of re-seating to eliminate identified seat leakage, correcting indication problems, or Inservice Inspection in MODES 3 and 4 provided:

1. *Only one valve in each train is cycled at a time; and*
2. *The same train redundant valve is verified closed and de-energized.*

In addition, the note associated with TRM Surveillance Requirement TSR 3.4.e.2 will be modified for consistency with the new TLCO note to permit post maintenance testing of repairs made in Modes 3 or 4. TSR 3.4.e.2 requires performing a complete cycle of each valve in the vent path from the control room every 18 months. This TSR is modified by a Note indicating that the surveillance shall not be performed in MODE 1, 2, 3, or 4. The TSR Note is being revised for consistency with the new TLCO Note to indicate:

This Surveillance shall not be performed in Mode 1, 2, 3, or 4 *except as allowed by TLCO Note.*

Reason for Activity:

The Reactor Coolant (RC) system reactor head vent valves are spring assisted-closed, pressure seated valves meaning the valves utilize system pressure to fully seat the valve plugs. The valves have occasionally experienced minor seat leakage during unit startup when system pressure is low. Typically, this leakage stops by the time the unit reaches normal operating pressure (maximum differential pressure across the valve). If the leakage is still present at normal operating pressure, the valve plugs can be reseated (cycled) in an attempt to achieve better valve closure. Currently, TRM TLCO 3.4.e does not permit cycling of RC system head vent valves in Modes 3 or 4.

The Braidwood ISI plan requires occasional pressure testing of the section of piping located between the inboard and outboard RC system head vent valves. The preferred method of performing this pressure test is to open the inboard valve when the unit reaches normal operating pressure. This TRM revision will permit this required ISI pressure test.

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

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: TRM TLCO 3.4.e **Revision Number: : TRM Change Request 05-004**

Title: Reactor Vessel Head Vents

This revision to TRM TLCO 3.4.e will eliminate the need for the Operations Department to make unnecessary Mode changes to address identified seat leakage. With this revision, seat leakage may be resolved without returning the unit to Mode 5 thus avoiding the challenges to plant equipment during Mode changes. This revision will also permit the required ISI related pressure tests of the piping associated with the head vent system.

This revision will not affect any design bases or safety analysis described in the UFSAR. The RC system head vent system is not credited in any transient or accident analysis. The system is designed and constructed in accordance with ASME Section III Class 1 requirements for operation at full RC system pressure and temperature so this TRM revision will not result in the operation of any SSC's beyond their design conditions.

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 TRM revision effectively alters the methods used to control a design bases function established in 10 CFR 50.46a that requires "...(2) There will not be an inadvertent or irreversible actuation of a vent.". The existing design relies on two in-series valves, controlled by separate control switches, to ensure a single failure would not result in an inadvertent or irreversible actuation. The TRM revision permits administrative control (de-energization) of an in-series valve to protect this design bases function while cycling the other valve. As such, a full 50.59 Evaluation was performed.

The Evaluation determined that Braidwood Station may implement the activity per plant procedures without obtaining a License Amendment. This conclusion was reached based on the fact that there are no physical changes to any SSC so all original design considerations, including potential failure modes, and the effects of SSC failure, remain valid. Since there are no physical changes to any SSC, the consequences of SSC failure remain bounded by the existing analyses. The revision to the TRM does not result in an SSC being operated beyond design limits therefore there are no new failure modes introduced. Based on the design of the RC system head vent valves, dc-operated, spring-assist close, system pressure seated, fail-closed, solenoid valves, the de-energization of one valve is considered to be equivalent to two in-series valves since the solenoid valve can not open and there is no concern of an inadvertent opening caused by switch failure or wiring short. Therefore, the design bases function remains protected and there is no increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

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.**	BRW-E-2005-81	**Rev.**	0

50.59 REVIEW COVERSHEET FORM

BRW-E-2005-143

Station/Unit(s): **Braidwood Station Unit 00**___________ Page 1 of 5

Activity/Document Number: EC #355995 ________________________________ **Revision Number: 00**__________

Title: Abandonment of the Make-Up Demineralizer System Caustic and Acid Injection for Regeneration of Resin in the Cation, Anion, and Mixed Bed Vessels

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 Abandonment of Caustic and Acid regeneration injection system for the Make-Up Demineralizers

Reason for Activity:
(Discuss why the proposed activity is being performed.)
The Unit 0 MUDS Caustic and Acid regeneration system has not been used since 1993. In that year, Braidwood Station changed from regenerating resin on –site to getting resin from a vendor off-site and performed regular resins change outs. The use of a reverse osmosis trailer unit has proven to be cost effective at eliminating the need for regenerating resin in the MUDS area. The vendor trailer removes the Total Dissolved Solids and Ions from the water via an EDI unit proving an effluent equivalent to the operating effluent requirements for the station. The Demineralizer vessels are downstream of this Reverse Osmosis unit and provide some back up to the vendor supplied trailer. The resin regeneration equipment has sat idle for several years and has not been abandoned per the applicable plant procedures CC-AA-109 and CC-MW-109-1001.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)
The abandonment of the Caustic and Acid regeneration injection system will not affect plant operations. Currently, all resin is regenerated off-site and the only function Braidwood performs is to change out the exhausted resin with regenerated resin. UFSAR, sections 9.2.3.2.1 and 9.2.3.2.2.5, discuss the use of the Caustic and Acid regeneration system to regenerate exhausted resin on the Make-Up Demineralizer system. There will need to be an update to these sections noting the abandonment of this equipment in the plant used for the regeneration of the resin in the Make-Up Demineralizer Vessels. The equipment described in these sections for regeneration of the resin in the Make-Up Demineralizer Vessels is going to be abandoned.

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 design functions of the Caustic and Acid Regeneration System is discussed in the UFSAR. This abandonment will involve the isolation of power to the 0WM04PA/B and the 0WM16PA/B eliminating the possibility of the regeneration system providing Acid or Caustic Chemical to the Demineralizer vessels. The method of changing out the exhausted resin with regenerated resin has been performed for several years and there is no plan to go back to regenerating resin on site. There is an Exempt Change (Reference CHRON 3120804; OSR 93-032), which declared the MUDS area as a TYPE 2 Exempt Change area. Therefore a 50.59 is not required for this particular change. The UFSAR will be update under ATI # 347835-01 DRP 11-030 to revise sections 9.2.3.2.1 and 9.2.3.2.2.5 due 07/29/05.

50.59 REVIEW COVERSHEET FORM

BRW-E-2005-143
Page 1 of 5

Station/Unit(s): **Braidwood Station Unit 00**

Activity/Document Number: EC #355995 **Revision Number:** **00**

Title: **Abandonment of the Make-Up Demineralizer System Caustic and Acid Injection for Regeneration of Resin in the Cation, Anion, and Mixed Bed Vessels**

Attachments:

Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)
Sections 9.2.3.2.1 and 9.2.3.2.2.5

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	00
X	**50.59 Evaluation**	**50.59 Evaluation No.**		**Rev.**	

BRW-E-2005-165

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: BwOR NAC-414 **Revision Number:** 0

Title: Braidwood NAC-LWT Cask Operating Procedure

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.)

BwOR NAC-414 is the station procedure that was developed based on the NAC procedure 315-P-02 Rev. 10. This procedure provides instructions to operate the NAC-LWT Cask at Braidwood Station. The purpose of this procedure is to: (1) provide the necessary steps associated with operating the NAC Legal Weight Truck (NAC-LWT) cask, including preparation for loading and unloading, actual loading or unloading, and preparation for shipment; (2) assist the cask user in preparing the necessary tools and services required for cask operation; and (3) acquaint the cask user with the operational features of the cask.

The NAC-LWT cask system consists of a cask body with closure lid, top and bottom impact limiters, a fuel basket, cask supports and tiedown, a lifting yoke and slings, and special tools for cask operation. For this particular application the capacity of the NAC-LWT cask is 25 PWR fuel rods. The gross shipping weight of the cask is approximately 52,000 pounds. In the shipping configuration, the combined weight of the cask and vehicle is less than the 80,000-pound limit for legal weight transportation. The cask body weighs approximately 43,420 pounds, is 199.75 inches long and has a maximum diameter of 44.22 inches (at the neutron shield expansion tank).

The NAC-LWT cask has been designed for ease of operation. The outer surface of the cask has been electropolished to minimize decontamination efforts by cask users. The closure lid and the two valve port covers are one-piece fixtures to speed removal and installation operations and thereby maintain personnel dose rates as low as reasonably achievable (ALARA). The closure lid has alignment grooves to facilitate remote placement. The valve port covers are provided for the vent and drain valves and are containment boundaries. The drain tube extends from the upper end forging region to the bottom of the cask cavity. The impact limiters and the personnel barrier are designed to be removed and installed without the aid of special lifting equipment.

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

Braidwood Unit 2 Cycle 10 and Braidwood Unit 1 Cycle 11 have both experienced fuel failures. Post irradiation exams have not determined the root cause. Both failed and non-failed rods are being shipped to a hot cell in Sweden for further examination in order to determine the root cause of the fuel failures.

The BwOR NAC-414 procedure will be utilized to operate the NAC-LWT cask, which will hold failed and non-failed fuel rods from Braidwood Unit 2 Cycle 10 and Braidwood Unit 1 Cycle 11.

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

The only accident in the UFSAR that is affected by the proposed activity is the Spent Fuel Cask Drop Accident in Section 15.7.5. The NAC cask will follow the path outlined in UFSAR Figure 15.7-1. Any single failure of any of the major components of the fuel handling crane would result in an undesirable condition, however, only a failure of the main hook would result in the dropping of the cask. This accident is expected to occur with the frequency of a limiting fault. The proposed activity does not modify the existing fuel handling crane. The NAC cask will be transported by attaching a lift yoke to the main hook. The yoke has two arms that attach to trunnions on the cask. Both the lifting trunnions and the lift yoke are designed against failure by the use of high design safety factors in the design and testing of the equipment in accordance with ANSI N14.6 and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Following fabrication and in accordance with the NAC maintenance program, the equipment is load tested to 300% of the design rated load followed by appropriate inspections and NDE. The lifting trunnions and the lift yoke have been designed and tested to preclude the possibility of an uncontrolled cask drop. Therefore, only a failure of the main hook would result in the dropping of the cask and the accident frequency would remain unchanged with the frequency of a limiting fault.

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

Activity/Document Number: **BwOR NAC-414** **Revision Number:** **0**

Title: **Braidwood NAC-LWT Cask Operating Procedure**

The proposed activity does not modify any SSC's at Braidwood. The NAC cask will follow the path outlined in UFSAR Figure 15.7-1. If a cask drop accident were to occur along this path, no SSC's important to safety would be caused to malfunction.

Main control room doses were calculated as part of the radiological analysis for the dropped NAC cask. The resulting control room doses were:

CR	5.6E-6	rem whole body
	1.3E-2	rem beta-skin
	2.1E-3	rem thyroid.

The calculated doses are well below the applicable acceptance limits of 5 rem whole body, 30 rem beta-skin and 30 rem thyroid. Therefore, main control room habitability will not be adversely affected by the proposed activity. In summary, the proposed activity will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

The proposed activity affects the radiological consequences of the Spent Fuel Cask Drop Accident in UFSAR Section 15.7.5. The current analysis assumes that appropriate impact limiting devices are employed during movement of the cask, the cask will not be damaged if it is dropped, and there will be no release of radioactive materials to the public. The NAC cask will be moved in the fuel handling building without impact limiting devices and the cask is subject to damage if dropped. Therefore, a radiological analysis was performed to determine offsite doses in the event that the NAC cask is dropped when loaded with the five fuel rods scheduled for shipment. The analysis determined that the event would result in the following doses:

EAB	4.6E-5	rem whole body
	6.2E-4	rem thyroid
LPZ	4.3E-6	rem whole body
	5.7E-5	rem thyroid

The acceptance criteria in NUREG 0800 for the Spent Fuel Cask Drop Accidents are 25% or less of the 10 CFR Part 100 exposure guideline values, i.e., 75 rem for the thyroid and 6 rem for whole body doses. An increase in consequences is defined to be no more than minimal if the increase is less than or equal to 10 percent of the difference between the current calculated dose value and the regulatory guideline value. Given that the current calculated dose values are zero, the new calculated dose values must be less than 7.5 rem for thyroid and 0.6 rem for whole body doses. The new calculated dose values are well below 7.5 rem for thyroid and 0.6 rem whole body. Therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

The proposed activity does not modify any SSC's at Braidwood. The NAC cask will follow the path outlined in UFSAR Figure 15.7-1. If a cask drop accident were to occur along this path, no SSC's important to safety would be caused to malfunction. Therefore, the proposed activity does not increase the consequences of a malfunction of an SSC important to safety and does not create the possibility for malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

The NAC cask will be brought to the pool area so as to enter the cask pit without passing over the spent fuel pool. Therefore, there is no possibility of dropping the cask into the spent fuel pool and damaging stored fuel assemblies. All design functions related to the cask pit will continue to be met with the insertion of the NAC cask into the cask pit. The center of gravity for the unloaded and loaded NAC Cask is approximately 99 inches measured from the bottom surface of the cask body. The cask pit is 132 inches wide by 132 inches long. If the NAC cask drops on the exterior spent fuel pool wall, and tips, it will land on the cask pit walls, which are designed to withstand the impact force due to a falling cask. The NAC cask will not fall outside the cask pit area since its center of gravity remains within the cask pit envelope and thus will not affect the fuel in the spent fuel pool. Therefore, the Fuel Handling Accident remains bounding with respect to damage of fuel stored in the spent fuel pool.

The proposed activity involves the loading of 5 fuel rods into a NAC cask for shipment offsite. The rods will be destructively tested and will never again be used in a reactor core. Therefore, the fuel cladding design bases parameters (DNBR, Centerline

50.59 REVIEW COVERSHEET FORM

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

Activity/Document Number: **BwOR NAC-414** **Revision Number:** **0**

Title: **Braidwood NAC-LWT Cask Operating Procedure**

fuel melting temperature, linear heat rate, fuel enthalpy, clad strain, fuel burnup, peak clad temperature, and cladding oxidation) are not exceeded or altered.

The proposed activity affects the method of evaluation for the Spent Fuel Cask Drop Accident in UFSAR Section 15.7.5. The current analysis assumes that appropriate impact limiting devices are employed during movement of the cask, the cask will not be damaged if it is dropped, and there will be no release of radioactive materials to the public. The NAC cask will be moved in the fuel handling building without impact limiting devices and the cask is subject to damage if dropped. Therefore, a radiological analysis was performed to determine offsite doses in the event that the NAC cask is dropped when loaded with the five fuel rods scheduled for shipment.

When the NAC cask is dropped, cask integrity is lost, and all five fuel rods are damaged. This results in the release of the fission product gap activity to the fuel building atmosphere. The activity release is limited to the iodines and noble gases. The source term was calculated using the ORIGEN2 computer code and actual power histories for the five fuel rods. Thyroid doses were calculated based on ICRP 30 dose conversion factors. Whole-body and beta-skin doses due to iodine releases were calculated based on ICRP 38 nuclide average disintegration energies. Whole-body and beta-skin doses due to noble gas releases were calculated based on R.G. 1.109 dose conversion factors. No credit was taken for filtration of activity releases provided by the fuel building ventilation system. The RADTRAD computer code was employed to calculate the activity release and resulting doses. RADTRAD is described in NUREG/CR-6604. Therefore, NRC approved methodology was used for the radiological dose calculation. The proposed activity does not result in a departure from a method of evaluation described in the UFSAR.

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.)

NRC approved methodology was used for the radiological dose calculation. Therefore, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR. The results of the radiological dose calculation did not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

In addition, the proposed activity at Braidwood conforms to Revision 39 of the Certificate of Compliance for the NAC-LWT cask package. The fuel rods being shipped are less than 5 weight percent U-235, have an active fuel length less than 150 inches, and have a pellet diameter less than 0.3765 inch. The burnup is less than 60,000 MWD/MTU and the cool time is greater than 150 days. Less than 25 rods are being shipped and the decay heat does not exceed 1.41 kilowatts. Since the cask will be loaded under water, the cask will be vacuum dried. The package will be leak tested prior to shipment.

This activity 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.**	**BRW-E-2005-165**	**Rev.**	**0**

BRW-E-2006-61

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Braidwood/ 01

Activity/Document Number: Letdown Booster Pump PMT/ SPP-05-011 **Revision Number:** 0

Title: LETDOWN BOOSTER PUMP 1CV03P POST MAINTENANCE TEST

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 Residual Heat Removal (RH) system functions as part of Emergency Core Cooling System (ECCS) during a Loss Of Coolant Accident (LOCA). During the post LOCA recirculation phase, the RH system delivers water from the containment ECCS sump to the Reactor Coolant (RC) system. The RH system pressure boundary functions as the radiological boundary to prevent contaminated ECCS sump water from leaking outside the containment. The normally locked closed RH isolation valves 1RH8703A/B provide the RH system pressure boundary function.

The proposed activity will open the normally locked closed isolation valves 1RH8703A/B that provides the RH system pressure boundary isolation point between the RH system and the letdown booster (LB) sub-system. The UFSAR section 5.4.7.2 describes the RH isolation valves as normally locked closed valve. The valves are opened to provide a suction source from the RWST via the A RH train to the LB pump sub-system. The RH valves are maintained in the open position to support the post maintenance testing (PMT) that fills, vents and operates the LB pump sub-system in recirculation mode. The RH valves are then restored to their normal locked closed position once the PMT is complete.

Reason for Activity:

The reason for the activity is to support the PMT of the LB pump sub-system. The test will ensure the LB pump is operational prior to the refueling outage.

Effect of Activity:

The normally closed RH isolation valves 1RH8703A/B provide the RH system boundary. Since the valves are locked closed, no operator or automatic actions are required to establish the RH system pressure boundary. The proposed activity opens the RH isolation valves. In order to restore the RH system pressure boundary the RH isolation valves will have to be manually closed. The action to close the valves are included in the PMT procedure and will be assigned to a dedicated operator located in the vicinity of the valves.

The proposed activity will not impact the ability of the RH system to provide a pressure boundary during the recirculation phase of LOCA. The PMT procedure provides sufficient guidance to ensure the valves are closed prior to post LOCA recirculation. A dedicated operator will be located at the valves and in direct communication with the main control room. In the event of an LOCA, the operator in the main control room can identify the LOCA accident from main control room alarms and immediately notify the local operator to re-establish the RH system pressure boundary by closing the RH isolation valves. In addition the local operator may close the valves upon the start of the 1B RH pump as indicated by local pressure gauges for the 1B RH pump. The valves are located in the general area of the containment spray pump and can be easily accessed and closed prior to establishing post LOCA recirculation flow path.

Summary of Conclusion for the Activity's 50.59 Review:

The performance of the PMT testing will affect the plant by aligning the LB pump sub-system to the RWST, which creates an A RH train LCOAR. When the PMT testing is complete, the LB pump will be restored to its normal stand by configuration.

The attached 50.59 Evaluation has concluded that opening the RH system pressure boundary valves 1RH8703A/B do not require prior NRC approval because it does not result in an increase in the frequency of occurrence or consequences of an accident or malfunction of an SSC previously addressed in the UFSAR. In addition the PMT procedure does not create an accident of a

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Braidwood/ 01

Activity/Document Number: Letdown Booster Pump PMT/ SPP-05-011 **Revision Number:** 0

Title: LETDOWN BOOSTER PUMP 1CV03P POST MAINTENANCE TEST

different type or different result or malfunction of an SSC of a different type or different result than previously evaluated in the UFSAR.

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.**	BRW-E-2006-61	**Rev.**	0

BRW-E-2006-74

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Braidwood Unit 1

Activity /Document Number: TRM Change #06-009 **Revision Number:** N/A

Title: Change In-Core Decay Time for A1R12

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

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

The proposed activity makes the following Technical Requirements Manual (TRM) changes to reduce the minimum required In-Core Decay Time (ICDT) for A1R12 from 100 hours to 70 hours:

- **Braidwood TRM Section 3.9.a, "Decay time," states "The reactor shall be subcritical for ≥ the last 100 hours (≥ 76 hours for A2R11)." This activity will revise this statement by replacing "(≥ 76 hours for A2R11)" with "(≥ 70 hours for A1R12)."**
- **Condition A under TRM 3.9.a states "Reactor subcritical for < 100 hours (< 76 hours for A2R11)". This activity will replace "(< 76 hours for A2R11)" with "(< 70 hours for A1R12)".**
- **Surveillance requirement TSR 3.9.a.1 will be revised by replacing "≥ 76 hours for A2R11" with "≥ 70 hours for A1R12".**

Reason for Activity:

Discuss why the proposed activity is being performed)

It is anticipated that during A1R12, 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 TRM fuel movement ICDT constraint of 100 hours after reactor shutdown.

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

The proposed changes will allow starting A1R12 Fuel offloading activities earlier than 100 hours.

The Byron and Braidwood spent fuel pool cooling design basis analysis is based on the minimum ICDT of 100 hours prior to starting fuel transfer, however an outage specific evaluation has been performed to support a reduced ICDT for A1R12. Starting core offload at 70 hours after shutdown will not result in increasing the design basis heat load for the Spent Fuel Pool (SFP) Cooling System.

Moving fuel early does result in an increase in the heat load input to the Spent Fuel pool compared to starting core offload at 100 hrs or later. However, the overall heat load to the spent fuel pool will not be greater than the design basis heat load, since the actual heat load in the SFP due to previously stored fuel assemblies is less than the heat load that was included in the SFP design basis analysis. The impact on the actual spent fuel pool temperature is expected to be minimal since two (2) trains of cooling can be operated during core offloads. Limitations and alarms

50.59 REVIEW COVERSHEET FORM

Activity /Document Number: TRM Change #06-009 **Revision Number:** N/A

Title: Change In-Core Decay Time for A1R12

related to Spent Fuel temperatures are in place during this activity. The limitation from EC #359062 for moving fuel at an average rate not greater than seven (7) assemblies per hour is not expected to have any impact on refueling operations, since during the last several refueling outages this offload rate has not been exceeded.

The current radiological design basis analysis for the Fuel Handling Accident is based on a minimum decay time of 48 hours prior to movement of irradiated fuel assemblies. As part of the Power Uprate Project, the radiological consequences of a Fuel Handling accident were evaluated and it was demonstrated that an ICDT of greater than or equal to 48 hours is acceptable for radiological considerations (Reference 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). Sections B3.9.4 and B3.9.7 of the Braidwood Technical Specification Bases are not being revised since the minimum ICDT for radiological considerations is not being revised and the revised ICDT for A1R12 still meets this constraint.

Occupational Radiation Dose

Occupational radiation dose will remain within limits. Per UFSAR tables 12.3-1 and 12.3-2, areas in the plant are divided 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 area affected by the defueling operation is designated as High Radiation area (Zone III). Access to these areas is controlled in accordance with station procedures and RWP. 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.

Summary of Conclusion for the Activities 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 does not increase the frequency of occurrence of a Fuel Handling Accident or a Loss of Spent Fuel Pool Cooling event, or increase the likelihood of occurrence of a malfunction of an SSC important to safety. This is based on the outage specific evaluation that concludes the total actual heat load in the Spent Fuel Pool as a result of the reduced ICDT is bounded by the total heat load specified in the design basis analysis.

The Fuel Handling Bldg radiation monitors and ventilation system are not adversely impacted. The monitors are not degraded by the radiation field expected due to the shorter ICDT. The likelihood of a malfunction of the Spent Fuel Pool Cooling system is not increased since the heat load on the system due to the reduced ICDT is bounded by the design basis analysis. Since the maximum bulk water temperature is not affected, the qualification of the spent fuel pool structure is not degraded.

This activity does not result in an increase in the consequences of an accident or in the consequences of a malfunction of an SSC important to safety. The offsite dose resulting from a Fuel Handling Accident with a minimum In-Core Decay Time of 70 hours is bounded by the design basis Fuel Handling Accident dose with a minimum ICDT of 48 hours.

50.59 REVIEW COVERSHEET FORM

Activity /Document Number: TRM Change #06-009 **Revision Number:** N/A

Title: Change In-Core Decay Time for A1R12

This activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR as there is no new equipment being introduced, and all existing fuel transfer equipment is being operated using existing procedures.

This activity does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR. The increase in heat load in the Spent Fuel Pool has been evaluated; although input parameters (offload start time and fuel transfer rate) have been changed, the resulting impact on the SFP bulk water temperature analysis is bounded by the design basis analysis. In addition, the local water temperature, fuel cladding temperature, and maximum heat flux have also been evaluated and have been found to be acceptable.

The adequacy of the reduced ICDT for A1R12 is based on the additional margin remaining in background decay heat since the SFP is not filled to its capacity.

The reduction in ICDT does not result in a change in the internal containment pressure that would represent a challenge to the containment design basis limit of 50 psig. The maximum cladding temperature for the spent fuel is well below the design basis limit of 2,200 °F. Therefore, the reduced ICDT does not result in exceeding design basis limits for a fission product barrier. In addition, this activity does not make any physical changes to the spent fuel, the containment or the RCS boundary that would result in altering their design basis limit.

.is activity does not change the method of evaluation for the Spent Fuel Pool Cooling System described in the UFSAR or in the SER for the Power Uprate Project. 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 design basis analysis.

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.**	BRW-E-2006-74	**Rev.**	0

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Braidwood Unit 0

Activity /Document Number: Temporary Configuration Change Package EC #358522, 358725 **Revision Number:** 1, 3

Title: Addition of Temporary Storage Capacity for Processed Liquid Radwaste

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

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves)
The proposed activity is a contingency to store processed radwaste water. Temporary storage tanks will be provided to accept processed wastewater from permanent plant release tanks 0WX01T and 0WX26T. The temporary storage tanks may be located inside and/or outside the permanent plant structure with a combined estimated storage capacity not to exceed 800,000 gallons. The waste water that will be transferred to the tanks will be sampled to ensure necessary steps are taken to meet the Braidwood Technical Requirements Manual (TRM) Appendix L limits for storage and Offsite Dose Calculation Manual (ODCM) limits for controlled release. The permanent pumps on the discharge of the Release tanks and rubber hoses will be used to transfer the fluid from the piping downstream of the release tanks to the temporary tanks. A temporary pump is used to transfer liquid radwaste between the outdoor tanks and those located in the old Steam Generator Replacement Project (SGRP) office building. When routed outside the permanent plant structure, the rubber hoses will be properly heat traced, when necessary, to minimize potential line freezing during cold weather. Heaters, and a recirculation pump when necessary, may also be used for the tanks located outdoors to prevent freezing. Heat tracing for the transfer hose and the tank nozzles, and the heaters for the tanks are addressed in EC358866 and 358798.

Additionally, a berm will be provided for tanks that may be located outside the permanent plant structure to contain any accidental liquid spilled to prevent it from escaping to the ground water. The berm is designed to contain the contents of one full tank plus a 6-inch allowance for rain/snow (Reference EC #358725). A berm is also provided for the tanks located inside the SGRP Office Building, designed to hold the contents of one tank.

Formal procedures control transferring the processed water from the discharge of the Release tanks to the temporary storage tanks (BwOP WX-501T4 and BwOP WX-526T4), transferring the contents of a temporary storage tank to another temporary storage tank (BwOP WX-601), and transferring the contents of a temporary storage tank to the release tanks (BwOP WX-600). Procedure EN-BR-402-0006 specifies the walkdown criteria for the temporary storage tanks.

Adequate floor drain system exists to accommodate any accidental spill drainage from tanks that may be located inside the permanent plant structure. The temporary tanks will be equipped with local level indication; the tank level will be monitored when being filled to ensure tanks do not over flow. The temporary storage tanks will not be used for direct liquid release to the environment.

Reason for Activity:
Discuss why the proposed activity is being performed)
Processed waste water stored in permanent plant release tanks 0WX01T and 0WX26T is released to the Kankakee river via the circulating water blowdown line described in UFSAR section 11.2.1.1. The proposed activity will provide temporary storage capacity should the station decide to delay water release to the river or process it by other means in the future.

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Effect of Activity:
(Discuss how the activity impacts plant operations, design basis, or safety analyses described in the UFSAR.)
The temporary storage of processed waste water will not adversely impact plant operations described in the UFSAR. The added storage tanks are considered "Outside temporary tanks" and thus fall within the requirements of the Braidwood Technical Requirements Manual (TRM) Appendix L, "Explosive Gas and Storage Tank Radioactivity Monitoring Program".

The combined radioactivity contents in the tanks will be limited to 10 Ci as allowed by TRM Appendix L. The radioactivity contents of an outside temporary tank is limited so that, upon an uncontrolled release from these tanks, the concentration at the nearest potable water supply and the nearest surface water supply in an unrestricted area, will be less than the limits of 10CFR20, Appendix B, Table 2, Column 2. The 10 Ci limit does not include tritium and dissolved or entrained noble gases. This exception is part of the Braidwood Licensing basis and is consistent with the requirements of NUREG-0133, "Preparation of Radiological Effluents Technical Specifications for Nuclear Power Plants". Any accidental release of processed water while being transferred or being stored is bounded by the analyses that support the TRM Appendix L limitations. Braidwood TRM Appendix L and the Braidwood ODCM implement the requirements of Braidwood Technical Specifications Section 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program".

Although the water will be stored within the temporary tanks, the water must be processed such that the contents of the temporary tanks are releasable to the Kankakee River. Water transfer to the temporary tanks will be controlled via station operating procedures. The procedures describe proper monitoring of the liquid transfer and temporary storage including sampling to ensure ODCM and TRM Appendix L limits continue to be met. As outlined in EC 358522 and EC 358725, steps are taken to ensure that any potential release from the temporary storage tanks is considered properly as part of the Braidwood Station Offsite Dose Calculation Manual (ODCM).

The monitoring and quantification of gaseous release is consistent with the ODCM requirements, and all releases will be ...ntified and included in the periodic reports for offsite releases. These actions ensure compliance with the requirements of Braidwood TRM Appendix A, "ODCM and Radiological Controls Reports and Program", and Appendix D, "Radioactive Effluents Controls Program". TRM Appendix L allows the added temporary storage tanks; thus no changes to the Braidwood TRM are needed.

It is important to note that, while the limit of 10 Ci has been applied to the contents of the tanks, the actual specific radioactivity contents are expected to be significantly lower than 10 Ci. The reason for this is that the water that is transferred to the temporary storage tanks has been processed through the liquid radwaste system and would have been ready for release to the Kankakee River. In fact, a review of past releases from the release tanks to the plant circulating water blowdown line shows, for a released volume comparable to the volume of a temporary storage tank (about 20,000 gallons) a total radioactivity of less than 500 microCi, excluding tritium and entrained or dissolved noble gases is typical. Thus, an uncontrolled release from the added temporary tanks is expected to result in concentrations significantly lower than the limits referenced in TRM Appendix L.

The berm around each tanks' location is designed to contain the contents of one tank. This is conservative as the TRM Appendix L limitation of 10 curies applies to unprotected tanks, which do not have a berm. Thus the main function of the berm is to contain incidental leakage from the tanks.

<u>Discussion of Regulatory Guide 1.143,</u> "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light Water Nuclear Power Plants, Revision 0"

The liquid radwaste system collects, monitors, and recycles or releases, with or without treatment where appropriate, all potential radioactive liquid waste produced by the station during normal operation and maintenance, as well as transient conditions. UFSAR Section 11.2.1.11 states that the liquid radwaste system meets the criteria of Regulatory Guide 1.143, Revision 0. For the purpose of the activities specified in EC 358522 and 358725, Braidwood Station takes exception to the requirements of RG 1.143.

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The table below provides a listing of the specific design requirements from Regulatory Guide 1.143 and the corresponding Braidwood design/configuration.

Regulatory Guide 1.143 Rev. 0 Requirements	Temporary Storage Tanks Design
For Atmospheric tanks, fiberglass reinforced plastic tanks may be used in accordance with appropriate Section 10, ASME Boiler and Pressure Vessel Code for application at ambient temperature.	RG 1.143 is not met for design, but it is exceeded for materials. The temporary tanks are atmospheric tanks and are of steel construction. Their design is judged more robust than that required by the Reg. Guide 1.143 Revision 0.
Piping and valves-ANSI B31.1	RG 1.143 is not met. The rubber hose that is used to transfer the liquid radwaste does not meet the requirements of RG 1.143. The ratings for this hose exceed expected operating conditions.
Foundation and structures that house the liquid radwaste system should be designed to the seismic requirements described in the Reg. Guide.	RG 1.143 is not met. The temporary equipment does not meet RG 1.143. In order to minimize the impact of this item, the cumulative radioactivity contents of the tanks will be limited to that allowed by TRM Appendix L.
Equipment and components used to collect, process, and store liquid radioactive waste need NOT be designed to the seismic criteria in the Reg. Guide.	RG 1.143 is met. The temporary tanks and associated piping / hoses are used to store liquid radioactive waste and therefore, are not required to be designed for seismic loads. The temporary tanks have been evaluated to meet the applicable wind loads.
All tanks should be designed to prevent uncontrolled release of radioactive material due to spillage (in buildings or from outdoor tanks).	RG 1.143 is met. The temporary storage tanks that are not located within a permanent plant structure including those located within the SGRP Office Building are provided with a berm. This berm is designed to contain the volume of one tank plus an allowance of 6-inches for rain water.
All tanks inside and outside the plant should have provisions to monitor liquid levels. Potential overflow conditions should actuate alarms.	RG 1.143 is not met, but controls provide equivalent protection. The temporary tanks have local level indication. Alarms for potential overflow conditions are not available, however, as part of the operating procedures, the tank level is monitored during filling activities, which minimizes the risk of tank overflow. The tanks are isolated after they are filled.
All tanks overflows and drains and sample lines should be routed to the liquid radwaste treatment system. Intermediate retention is acceptable.	RG 1.143 is met as intermediate retention is provided. Inadvertent spillage from tank overflows, drains and sample lines are routed into the bermed area. Walkdown by Operations personnel will check for leakage from the tanks and connected equipment. Radiation Protection will ensure proper disposal of the leakage volume.
Indoor tanks should have curbs or elevated thresholds with floor drains routed to the radwaste system.	RG 1.143 is not met. The only temporary tanks that are housed internally to a structure are those within the SGRP Office Building and those within the Turbine Building. A failure of a tank in the Turbine Building would ultimately drain to the fire and oil sump, a monitored path. The tanks in the SGRP Office Building are provided with a berm that is sized to hold the contents of one tank.
Outdoor tanks should have a dike or retention pond capable of preventing runoff in the event of a tank overflow and should have provisions for sampling collected liquids and routing them to the liquid radwaste system.	RG 1.143 is not met. Liquid transfer to outdoor tanks is attended and monitored when a tank is filled to minimize the potential for tank overflow. Spillage volume inside the bermed area is sampled for proper disposal.
Piping systems should be hydrostatically tested at no less than 75 psig. The test pressure should be held for a minimum of 30 minutes, with no leakage indicated.	RG 1.143 is not met. Liquid transfer to each tank does not utilize hard piping. Temporary transfer hoses are made of reinforced rubber. The ratings for these hoses exceed the expected operating conditions. In order to minimize the potential for leakage, all new connections are verified to be leak-free prior to each transfer. An additional leak check is performed with the temporary system components in operation. Finally, a liquid transfer is stopped should leakage develop.
Screwed connections for process pipe should not be used except for instrumentation connections.	RG 1.143 is not met. Temporary tanks isolation valves and non-metallic transfer hoses have threaded connections. Administrative guidance has been provided in the design package to wrap these connections and use catch containers as required to mitigate leakage/runoffs.
Establish a quality assurance program sufficient to ...ure that all design, construction, and testing ...visions are met.	Exceed RG 1.143. Design and installation of the TCCP is performed in accordance with approved procedures prepared in accordance with the Exelon Quality Assurance Program. The Exelon Quality Assurance Program meets the requirements of 10CFR50, Appendix B.

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The fact that the addition of the temporary storage tanks does not fully meet Regulatory Guide 1.143 is deemed to be acceptable because the design of the temporary equipment, the related procedures and controls, and the limitation on the cumulative radioactivity contents in the tanks collectively support meeting the requirements of 10CFR50 Appendix A, Criterion 60, 61, 63 and 64.

- 10CFR50 Appendix A, Criterion 60, Control of Releases of Radioactive Materials to the Environment
  The temporary storage tanks are needed to provide additional storage capacity when the normal water releases to the environment, via the Circulating Water Blowdown line to the Kankakee River, are not performed. Provisions are in place to account for any potential gaseous release from the tanks in the Station ODCM. Liquid release, when performed, is by returning the temporarily stored water back to the release tanks for controlled release.

- 10CFR50 Appendix A, Criterion 61, Fuel Storage and Handling and Radioactivity Control
  The water that is stored within the temporary tanks is processed to remove radionuclides such that the contents of the temporary tanks are releasable to the Kankakee River. As a result, shielding is not required due to the low level of radionuclides present within the stored water. Provisions are in place to control / quantify gaseous releases from the tanks and contain / cleanup inadvertent liquid spills.

- 10CFR50 Appendix A, Criterion 63, Monitoring Fuel and Waste Storage
  Administrative procedures sample the water prior to introduction into temporary storage to ensure ODCM and Braidwood TRM Appendix L limits are met and periodic sampling ensures these limits continue to be met. Also the contents of the temporary tanks are monitored so that any potential release is accounted properly.

- 10CFR50 Appendix A, Criterion 64, Monitoring Radioactive Releases
  The contents of each tank are monitored and any potential gaseous release is quantified as part of this activity. Liquid release, when performed, is by returning the temporarily stored water back to the release tanks for controlled release. The cumulative radioactivity in the temporary tanks is limited to the requirements of TRM Appendix L.

Since the activity addressed by this evaluation is temporary, the UFSAR will not be revised to reflect the addition of the temporary storage tanks or to reflect the position in relation to Regulatory Guide 1.143.

Summary of Conclusion for the Activities 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 does not need NRC approval prior to implementation.

The added temporary storage tanks fall within the storage tank radioactivity monitoring program from Braidwood TRM Appendix L.

This activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR because it does not increase the frequency of occurrence of a failure of an outdoor, unprotected tank or a failure of the Recycle Hold-up tank. This activity does not increase the likelihood of occurrence of a malfunction of an SSC important to safety. The added components will not expose the Liquid Radwaste system (WX) equipment/components to conditions that are beyond their design basis. The temporary tanks are connected to the WX system only during the filling process.

Concurrent Failure of all the added temporary tanks would be necessary to result in an event that is equivalent to failure of one outdoor temporary tank that is limited to the radioactivity contents specified in TRM Appendix L. The added storage tanks do not affect the failure frequency for the RHUT.

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This activity does not result in an increase in the consequences of an accident or in the consequences of a malfunction of an SSC important to safety. The offsite dose resulting from a recycle hold-up tank failure (i.e., an atmospheric release) bounds the offsite dose resulting from an improbable failure of all the temporary tanks for atmospheric release. For ground liquid release, limiting the radioactivity contents in the tanks to 10 Ci or less ensures that the consequences of a liquid release are bounded by the consequences of a failure of a temporary outdoor tank as specified in TRM Appendix L, which is consistent with the requirements of NUREG-0800, 15.7.3 "Postulated Radioactive Releases due to Liquid-Containing Tank Failures", section III.4.

This activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR as failure of temporary outdoor tanks is addressed as part of TRM Appendix L.

This activity does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR. Results of tank failure are bounded by failure of a Recycle Holdup tank for atmospheric release and by failure of an unprotected tank for ground release as addressed in TRM Appendix L. This activity does not result in changes to the fuel clad, Reactor Coolant System pressure boundary piping or the containment building. This activity does not change the method of evaluation for the failure of a Recycle Holdup tank for atmospheric release or an unprotected outdoor tank for ground release as addressed in TRM Appendix L.

Although this activity does not fully meet the requirements of RG 1.143, the design of the temporary equipment, the related procedures and controls, and the limitation on the cumulative radioactivity contents in the tanks collectively support meeting the requirements of 10CFR50 Appendix A, Criterion 60, 61, 63 and 64 and is bounded by existing analyses.

Although tritium is not included in the radioactivity contents limitation for the temporary tanks, the actual tritium concentration in each tank is recorded as required in operating procedures BwOP WX-501T4 and BwOP WX-526T4. In the unlikely event of an uncontrolled liquid release from the tanks, this information will be used to quantify curies of tritium released from the tanks. This unlikely release would be reported as part of the Radioactive Effluent Release Report, which is required to be submitted to the NRC per the requirements of Technical Specification Section 5.6.3, "Radioactive Effluent Release Report".

Furthermore, consequences from a water spill from the temporary tanks would be limited because remedial actions would be taken to minimize/prevent migration of the spilled water to the groundwater near the site.

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.	BRW-E-2006-88	Rev.	0