Exelon.[SM]
Nuclear

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December 16, 2004
BW040111

10 CFR 50.59(d)(2)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

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, 2002, through June 18, 2004, and consists of the coversheets for changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR), and tests or experiments not described in the UFSAR.

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

Respectfully,

T. P. Joyce

T. P. Joyce
Site Vice President
Braidwood Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station

IE47

Report of Evaluations

03-Dec-04

Total Records = 13

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-2002-234 Revise control switch operation for the RHR miniflow isolation valves, to allow maintained contact for Open position.	9/9/2002	Yes	1/2	NELSON, GARY W. MOD DESIGN	KON, MICHAEL	Yes	10-001	RH	1/2RH610, 1/2RH611	Modifications ECs 337450, 337451, 337452& 54	
BRW-E-2002-253 As built DCR EC 337189 for 0C WS pump cooling water Y-strainer drain valve and piping which do not appear on drawing M-43-1	9/5/2002	Yes	0	MATTHEWS, JOHN L. PLANT ENGINEERING	MULLINS, SAMU	No		WS	0WS346	Drawing Change 337189	
BRW-E-2002-300 Core Recriticality During Hot Leg Switchover	1/15/200:	Yes	1/2	STEVENS, TYRONE L BRD REACTOR ENGINEERING	STEVENS, TYRO	Yes		RX	N/A	UFSAR Revision	
BRW-E-2002-301 Replace SG blowdown hotwell condenser pump variable speed motor controllers with digital devices. Evaluation required due ONLY to being a digital replacement.	11/5/2002	Yes	1/2	NELSON, GARY W. MOD DESIGN	KON, MICHAEL	No		SD	1/2SD01JA, JB	Modifications EC 333913, 336394	
BRW-E-2003-5 This Evalauation was never completed. See Screening BRW-S-2003-85. Update of B/B UFSAR regarding description of the environmental qualification program and/or environmental data for plant areas as a result of Calc. BRW-01-0153-E/BYR01-068.	1/6/2003	Yes	1/2	SPISAK, MICHAEL J. MOD DESIGN	FERMIER, REX W	Yes	10-012	None	Various EQ Components	UFSAR Revision DRP 10-012	Calc. BRW-01-0153-E/BYR01-068 and supporting documents referenced therein.
BRW-E-2003-46 Reduce Incore DecayTime for A1R10 TRM Change #03-001	2/20/200:	Yes	1	PANICI, JOHN	RADICE, DANIEL	No		FC		Other TRM Change #03-001	
BRW-E-2003-93 Changes to procedures 1/2BwEP-0, 1/2BwEP-1, 1/2BwEP ES-1.2,1/2BwEP ES-1.3. Compensatory Action to support operation of SVAG valves post-Accident.	4/8/2003	No	1/2	PANICI, JOHN MOD DESIGN	PANICI, JOHN	No		SI		Other Procedure Changes	
BRW-E-2003-94 Revise 1(2)BwEP-0 and 1(2)BEP 0 to require manual shut down of the VV, VL and VW at Braidwood and the VV system at Byron following an ESF safety injection signal.	4/8/2003	No	0	RIEDINGER, DARREL MOD DESIGN	PANICI, JOHN	No		VV		Procedure Revision 1BwEP-0 r 101, 2BwEP-0 rev 103	

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-2003-148 Update UFSAR to Address NSAL-03-01	6/13/2003	No	1	STEVENS, TYRONE L BRD REACTOR ENGINEERING	NIEDERER, RON	No		None	N/A	UFSAR Revision 10-028	
BRW-E-2003-220 UFSAR Update addressing the revised structural component criterion (Rev 1 to Addendum 1 of WCAP-12488-A	9/15/2003	Yes	1/2	STEVENS, TYRONE L BRD REACTOR ENGINEERING	ROTHENBUEHLE	No		RX1		UFSAR Revision 10-043	
BRW-E-2003-229 TRM Change #03-015 Change minimum Incore Decay Time for A2R10 from 100 hrs to 65 hrs. (TRM Section 3.9.a)	9/22/2003	Yes	2	PANICI, JOHN MOD DESIGN	RADICE, DANIEL	No		FC	1/2FC01A	Other TRM Change #03-015	
BRW-E-2003-218 AMAG Implementation Using FW Header Flow	1/28/2004	Yes	1/2	WUNDER, ROBERT G MOD DESIGN	SHAH, MAHENDR	No		FW		Modifications 342431 and 344502	BRW-S-2003-109 and BRW-S-2003-210
BRW-E-2004-132 Canceled - not used	6/18/2004	No		KOENIG, ROBERT D		No					

BRW - E - 2002 - 234

50.59 REVIEW COVERSHEET FORM

Station: Braidwood Unit 1 and 2

Activity /Document Number: ECs 337450, 337451, 337452, 337454 **Revision Number:** 0

Title: Revised Control Switch Operation of RHR Miniflow valves 1(2)RH610 / 611

Description of Activity:

Change the operator mechanism of the control switches for the RHR miniflow line isolation valves 1(2)RH610/611 (the miniflow valves) to enable a maintained Open position. This will be a change from its original configuration of spring return to center from both Closed and Open positions. The spring return to center from Closed position will not be changed. The change will be done at the main control board 1(2)PM06J.

Reason for Activity:

The miniflow line provides a recirculation path for preventing deadheading of the RHR pumps on low flow conditions. Control switch maintained Open capability will allow operators to perform the required gradual heat up of the RHR pumps for a shutdown cooling start without requiring them to hold the switch Open for an extended time and/or until the valve's circuit breaker is de-energized. When an RHR pump is started, the MOV is normally open due to low discharge flow. The valve auto closes when discharge flow exceeds approximately 1400 gpm. This flow value is generally reached sooner in time than an adequate RHR pump warmup can occur when started for the shutdown cooling mode. Operating experience has shown that pump damage can occur if a slow controlled heatup is not performed when placing the RHR system in service for shutdown cooling. Maintaining the miniflow MOV open allows the controlled pump heatup to occur.

Effect of Activity:

This change will have the ultimate result to help preclude potential RHR pump damage when placed in service for shutdown cooling. It will directly alleviate a cumbersome procedural activity for operators during RHR pump manual starts. Presently, operators must hold the control switch Open until the valve's circuit breaker can be manually de-energized. This configuration change introduces no more than a minimal increase in probability that the miniflow line MOV could inadvertently be mispositioned. The miniflow valves are assumed to have their control switches in the automatic position to assure that injection flow to the vessel is not diverted or bypassed during an accident. The auto position will be the normal control board configuration. This change activity will not impact the capability of the RHR pump to start and provide ECCS flow when actuated by an automatic start signal.

Summary of Conclusion for the Activities 50.59 Review:

This evaluation concludes that the change activity can be implemented without prior NRC approval.

Attachments:

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

50.59 REVIEW COVERSHEET FORM

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-2002-234	**Rev.**	0
	50.59 Validation	**50.59 Validation No.**		**Rev.**	

BRW-E-2002-253

50.59 REVIEW COVERSHEET FORM

Page 1 of 9

Station: Braidwood

Activity/Document Number EC #337189 **Revision Number: 0**

Title: ASBT DCR - 0C WS pump Y-strainer drain valve and drain valve pipe

Description of Activity:
To process an As – Built (ASBT) DCR to show the installed configuration of a drain valve and associated piping for the 0C Nonessential Service Water (WS) pump motor and pump cooling water strainer 0WS01MC. In the field, the following is attached to the drain connection on Y-Type strainer 0WS01MC: 1/2" x 1-3/4" long nipple, 1/2" 45 deg elbow, 1/2" x 1-1/2" long nipple, 1/2" globe valve and a 1/2" x 2" long nipple (See Page 8). As part of the "As-Built" drawing change, the valve has been assigned valve number 0WS346 (Y-Strainer drain valve) and the 1/2"line has been assigned line number 0WSW5AC (Y-Strainer drain valve pipe). The new valve and line will be added to drawing M-43 Sheet 1 which already shows the Y- Strainer 0WS01MC. The Y-strainer filters the cooling water before it enters the motor cooler and pump bearings. Additionally, a new note (Note 7) will be added to the drawing which states, "THREADED PIPE CONNECTIONS ARE ACCEPTABLE FOR THE COOLING WATER Y-STRAINER DRAIN VALVE AND PIPING." The applicable Piping Design Table 100BB requires welded fabrication unless otherwise designated on the design drawings.

The 0A and 0B WS pumps have a similar drain valve and piping which already are shown on drawing M-43-1. After the drawing change is processed, the configuration in the field and on the drawing will be the same for all three WS pumps. The pipe and line numbers are 0WS340 and 0WSW5AA-1/2 for the 0A pump and 0WS342 and 0WSW5AB-1/2 for the 0B WS pump. These lines are also assembled with threaded pipe connections.

Drawing M-43-1 is UFSAR Figure 9.02-01, Sheet 1.

Reason for Activity:
To resolve the discrepancy between the field, drawing M-43-1 and UFSAR Figure 9.02-01, Sheet 1. The drain valve is necessary to maintain the pressure boundary of the strainer and allow for draining and flushing of the strainer. Therefore, the drawing is being changed rather than removing the valve and piping from the plant. The discrepancy was documented in Condition Report (CR) 105427 and the drawing change is being tracked by Corrective Action Program (CAPSYS) tracking item 105427-04.

Effect of Activity:
There is no safety significance to this change. The associated cooling water piping as well as the WS system itself is Non-safety/Non-seismic Category II and non-ASME Class D. The drain valve is normally closed and only operated during strainer draining and flushing operations. The only affect will be to add the drain valve 0WS346 to the WS mechanical line-up procedure BwOP WS-M0. The Y-strainer drain valve (0WS346) is connected to the drain valve pipe (0WSW5AC) where the drain valve is in the normally closed position during plant operation. The Y-strainer drain valve for the 0B pump (0WS343) and 0A pump (0WS340) have the same configuration as the drain valve for the 0C pump. The Y-strainer drain valve and drain valve piping Safety Category II (Non-Safety Related) and Non-ASME Section III (Quality Group D).

Summary of Conclusion for the Activities 50.59 Review:
A Screening per LS-AA-104-1003 indicates that a 50.59 Evaluation would not be required because there is no change being made to the UFSAR described function, however, "BwAP 1205-16, BRAIDWOOD STATION DRAWING TO UFSAR/FPR FIGURE CROSS-REFERENCE LISTING" requires a 50.59 Evaluation any time a UFSAR Figure is impacted by a drawing change. Condition Report 117650 has been generated against the discrepancy between BwAP 1205-16 and the 50.59 Resource Manual and ATI 117650-04 is tracking the revision to BwAP 1205-16.

Attachments:
Attach completed Applicability Review if 50.59 Screening is not required.

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

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☐	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	BRW-E-2002-253	**Rev.**	00
☐	**50.59 Validation**	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

Station: Byron / Braidwood
Activity/Document Number DRP 10-008 **Revision Number:** 0

Title: Core Recriticality During Hot Leg Switchover (NSAL-94-016, Revision 2, March 2002

Description of Activity:

Activity is to accept the analysis and update the UFSAR to take credit for the boron equivalent worth of xenon and control rod insertion to address the Post-LOCA Recriticality during hot leg switchover issue identified in NSAL-94-016 Revision 2.

Reason for Activity:

Westinghouse identified a potential safety issue regarding plant operation after a large cold leg break LOCA. The concern is that during hot leg switchover the core will be flushed with diluted sump solution, which may cause the core to return to criticality. The sump solution would be diluted since boron is assumed to accumulate in the core during the cold leg recirculation phase due to core boiling. The accumulation of boron in the core prevents the boron from being returned to the sump, which leads to a diluted sump solution. This issue is described in the following documents:

(1) NSAL-94-016, Core Recriticality During LOCA Hot Leg Recirculation, 7/25/94.
(2) NSAL-94-016, Supplement 1, Core Recriticality During LOCA Hot Leg Recirculation, 03-28-96.
(3) NSAL-94-016, Revision 1, Core Recriticality During Hot Leg Switchover, 03-30-99.
(4) NSAL-94-016, Revision 2, Core Recriticality During Hot Leg Switchover, 03-18-02

Currently both Byron and Braidwood have addressed this issue by assessing that, while some dilution of the sump boron concentration might occur in the hours between cold leg recirculation and hot leg switchover due to boil off and boron concentrating in the core, the additional water from the RWST following cold leg recirculation and Xenon credits at the time of hot leg switchover compensate for the sump dilution. Furthermore, the post-LOCA criticality check at cold leg recirculation which assumes all-rod-out, no xenon, and cold (68 to 212 °F) conditions (current licensing basis requirements) is still judged to be more limiting than at hot leg switchover. (For detail see Memo CAE-00-202/CCE-00-200). Also, as a defense in-depth, Memo CAE-00-202/CCE-00-200 further states that control rod insertion may be credited as an additional mitigating factor to address the issue. Currently the Byron and Braidwood licensing bases do not credit control rod insertion to address this issue.

NSAL 94-016, Revision 2, provides a generic evaluation that credits xenon credit and control rod insertion to conclude that the check of subcriticality at beginning of cold leg recirculation adequately addresses potential recriticality due to sump dilution entering hot leg recirculation provided that the initial subcriticality check does not credit control rods or xenon. Based on the work recently completed by WOG (WCAP - 15704) the licensing basis for Byron and Braidwood will now credit the boron worth of inserted control rods and Xenon credit at the time of hot leg switchover to address the potential for recriticality due to sump dilution when realigning to hot leg recirculation.

Effect of Activity:

Credit for control rod insertion will be taken to address the post-LOCA core recriticality issue identified in NSAL-94-016, Revision 2. Accordingly a UFSAR post-LOCA section will be added. See DRP 10-008 for details.

Summary of Conclusion for the Activities 50.59 Review:

This activity does not result in a departure from a method of evaluation described in the UFSAR. The UFSAR does not mention the details of the methodology for ensuring post-LOCA subcriticality. The connection to the UFSAR is through reference to licensed Westinghouse ECCS Methodology reported in WCAP-8339 and WCAP-8471-P-A. Note that neither WCAP-8339 nor WCAP-8471-P-A discusses specific post-LOCA requirements but only states that control rods are not modeled in the Westinghouse Large Break LOCA ECCS methodology. A post-LOCA section will be added in the UFSAR that includes a statement that the method for addressing the possibility of recriticality upon entering hot leg recirculation due to sump dilution will credit control rod insertion.

The NRC has previously approved this methodology (control rod insertion following a cold leg break to address post-LOCA recriticality issue) for D.C.Cook. The applicability of this methodology for By/Br is described in WCAP-15704. A screening and evaluation have been completed for this activity. The conclusion of the evalaution was that the activity may be implemented without prior NRC review and approval.

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

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☒	**50.59 Screening**	**50.59 Screening No.**	**6E-02-0249 / BRW-S-2002-299**	**Rev.**	**0**
☒	**50.59 Evaluation**	**50.59 Evaluation No.**	**6G-02-0014 / BRW-E-2002-300**	**Rev.**	**0**
☐	**50.59 Validation**	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

Station: Braidwood Unit 1, 2

Activity /Document Number: ECs 333913, 336394 **Revision Number:** 0

Title: Replace Steam Generator Blowdown Condenser Hotwell Pump Motor Controllers

Description of Activity:

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

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

Reason for Activity:

The intent of these changes is to improve the performance and reliability of the controllers and to reduce maintenance expenses.

The existing controller components are experiencing increased failure rates due to age degradation and are also becoming obsolete. Servicing and replacing available parts is not economical compared to installing a newer style controller. The most economical replacement is a commercially manufactured digital controller and has been recommended by Braidwood's servicing vendor.

fect of Activity:

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

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

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

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

Summary of Conclusion for the Activities 50.59 Review:

50.59 REVIEW COVERSHEET FORM

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

Although different components susceptible to failure are being installed, the result and consequences of a malfunction or failure are not different than previously analyzed in the UFSAR and are therefore bounded by existing UFSAR analyses. In addition, based on a technical review of new components and construction, Engineering has determined that there will not be more than a minimal increase in the likelihood of a malfunction, initiation (frequency) of a transient, or the likelihood of an accident.

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

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

References:

UFSAR
3.0 - Design of Structures, Components, Equipment, and Systems
8.0 - Electric Power
10.4.8 - Steam Generator Blowdown system
  10.4.8.2 System Description and Operation
  10.4.8.3 Safety Evaluation
11.2 Liquid Waste Management System
  11.2.2.1 Blowdown System
  Figure 11.2-1 Liquid Radwaste Processing System
15.0 - Accident Analyses

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

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	50.59 Screening No.		Rev.	
X	50.59 Evaluation	50.59 Evaluation No.	BRW-E-2002-301	Rev.	0
	50.59 Validation	50.59 Validation No.		Rev.	

BRW-E-2003-46

50.59 REVIEW COVERSHEET FORM

Station: Braidwood Unit 1

Activity /Document Number: TRM Change #03-001 **Revision Number:** N/A

Title: Change In-Core Decay Time for A1R10

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 A1R10 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 (≥ 72 hours with average water temperature of UHS ≤ 100 °F for A1R09)." This activity will revise this statement by replacing "(≥ 72 hours with average water temperature of UHS ≤ 100 °F for A1R09)" with "(≥ 65 hours for A1R10)."**

- **Condition A under TRM 3.9.a states "Reactor subcritical for < 100 hours (< 72 hours for A1R09)". This activity will replace "(< 72 hours for A1R09)" with "(< 65 hours for A1R10)".**

- **The proposed activity will delete Condition B in its entirety.**

  Condition B states "UHS average water temperature > 100 °F (A1R09 only)". A maximum UHS water temperature of 100 °F supports the current Spent Fuel pool water temperature evaluation. The limitation on UHS average water temperature was required due to an Interim Request for a Technical Specification Change to raise the maximum allowed UHS temperature from 100 °F to 102 °F. This interim request was in review by the NRC in the time period preceding core offloading for A1R09 but has since been withdrawn.

- **Surveillance requirement TSR 3.9.a.1 will be revised by replacing "≥ 72 hours for A1R09" with "≥ 65 hours for A1R10".**

- **Surveillance requirement TSR 3.9.a.2 will be deleted. This surveillance verified that the UHS average water temperature was ≤ 100 °F for A1R09. As discussed above, this surveillance is no longer required.**

The following procedures will be revised to remove the verification of the UHS temperature and any reference to TSR 3.9.a.2:

- **1BwGP 100-6T5, "Core Alteration Checklist", Revision 5**
- **1BwOS TRM 3.9.a.2, "Ultimate Heat Sink Temperature Surveillance", Revision 0**
- **1BwOL TRM 3.9.a, "Technical Requirements Manual (TRM) LCOAR- Decay Time- TRM TLCO 3.9.a", Revision 3**
- **1BwOSR 0.1-6, "Unit One–Mode 6–Shiftly and Daily Operating Surveillance", Revision 4**

50.59 REVIEW COVERSHEET FORM

Activity /Document Number: TRM Change #03-001 **Revision Number:** N/A

Title: Change In-Core Decay Time for A1R10

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

It is anticipated that during A1R10, 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 A1R10.

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. 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 A1R10 still meets this constraint.

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 A1R10 Fuel offloading activities earlier than 100 hours. This will save time on the critical path during the outage.

<u>Occupational Radiation Dose</u>
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 A1R10.

50.59 REVIEW COVERSHEET FORM

Activity /Document Number: TRM Change #03-001 **Revision Number:** N/A

Title: Change In-Core Decay Time for A1R10

Page 3 **of** 10

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 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 qualified to function in a radiation field that is higher than the calculated maximum does rate 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 a malfunction of an SSC important to safety. The 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 temperature, and maximum heat flux have also been evaluated and have been found to be acceptable.

The adequacy of the reduced ICDT for A1R10 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 the design basis analysis.

50.59 REVIEW COVERSHEET FORM

Activity /Document Number: TRM Change #03-001 **Revision Number:** N/A

Title: Change In-Core Decay Time for A1R10

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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-2003-46	**Rev.**	0

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50.59 REVIEW COVERSHEET FORM

Station: *Byron/Braidwood*

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

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

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

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

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

This evaluates the following procedure revisions:

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

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

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

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

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

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

50.59 REVIEW COVERSHEET FORM

Station: *Byron/Braidwood*

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

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

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

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

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

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

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

50.59 REVIEW COVERSHEET FORM

Station: *Byron/Braidwood*

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

Revision Number: See below

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

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

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

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

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

Forms Attached: (Check all that apply.)

[ ]	**Applicability Review**				
[ ]	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
X	**50.59 Evaluation**	**50.59 Evaluation No.**	*Byr 6G-03-0001* *Brw-E-2003-93*	**Rev.**	*1*

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50.59 REVIEW COVERSHEET FORM

LS-AA-104-1001
Revision 1

50.59 Evaluation No.: BRW-E-2003-94 / 6G-03-0002 **Rev. No.:** 0.0 **Page 1 of 5**

Station: Braidwood/Byron Units 1 and 2

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

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

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

Description of Activity:

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

Reason for Activity:

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

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

Effect of Activity:

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

Summary of Conclusion for the Activity's 50.59 Review:

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

50.59 REVIEW COVERSHEET FORM

50.59 Evaluation No.: BRW-E-2003-94 / 6G-03-0002 **Rev. No.:** 0.0

Station: Braidwood/Byron Units 1 and 2

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

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

Forms Attached: (Check all that apply.)

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

BRW-E-2003-148

Station: Braidwood Unit 1/Byron Unit 1

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

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

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

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

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

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

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

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

There is no effect on plant operations.

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

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

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

50.59 REVIEW COVERSHEET FORM

Station: Braidwood Unit 1/Byron Unit 1

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

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

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

Forms Attached: (Check all that apply.)

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

BRW-E-2003-218

50.59 REVIEW COVERSHEET FORM

Station: Byron/Braidwood Units 1&2

Activity/Document Number: AMAG Implementation Using FW Header Flow **Revision Number:** 3

Title: Safety Evaluation for the Implementation of the AMAG Correction Factor

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).

BACKGROUND

The Byron DCPs 344500 and 344518, and Braidwood DCP 346235, installs non-intrusive ultrasonic feedwater flow instrumentation on the feedwater supply common header 1(2)FW03B-30" in the turbine building. The ultrasonic transducers are mounted to brackets that are bolted to the feedwater piping. Separate 10 CFR 50.59 Safety Evaluation Screening (6E-03-0121) for Byron, and CC-AA-103 Attachment F, "Screening Criteria for Commercial Changes" for Braidwood, were developed to document the physical installation of the ultrasonic flow instrumentation.

This 10 CFR 50.59 Evaluation justifies using the ultrasonic flow measurements in the calorimetric calculations, and the revision or development of the procedures listed herein. The calorimetric calculations use a correction factor in the determination of the reactor thermal power for both the plant computer and the daily calorimetric calculations. The software for the plant computer has already been modified to accept this correction factor. The calorimetric software for this application has been Verified and Validated in accordance with Digital Technology Systems Quality Assurance (DTSQA) procedure IT-AA-101.

During reactor operation, discrete reactor power levels are determined on a continuous basis by the plant computer based on inputs received from various monitoring instruments. Of particular importance to the calorimetric calculations is feedwater flow. The current calorimetric calculation uses the feedwater flow as obtained from the feedwater venturis (FE-510, 520, 530 and 540). The feedwater flow rate is a function of the differential pressure across the venturi as measured by the flow transmitter. Industry and plant operating experience has shown that during operation fouling of the venturis may occur. Fouling of a venturi results in a reduction of the flow cross-section through the throat of the venturis, and an increase in the pressure drop through the throat due to the roughness of the fouling scale. Fouling and other induced measurement inaccuracies contribute to an indicated feedwater flow higher than actual. Indicated feedwater flows that are higher than actual values result in an overly conservative calorimetric reactor power calculation. The permanently installed ultrasonic transducers are connected to a shared data acquisition/data analyzer used to collect and analyze ultrasonic flow measurements. Correction factors are developed to relate the flow as indicated by the feedwater venturi, to that obtained from the ultrasonic measuring system installed on the feedwater header. To address this overly conservative calorimetric reactor power calculation, the correction factors are manually entered into the plant computer and daily calorimetric calculations to compensate for the feedwater venturi fouling and other venturi induced measurement uncertainties. The NRC has issued an SER on March 20,2000 accepting CE topical report CENPD-397P Rev. 01 for the use of cross flow ultrasonic flow measurement to correct feedwater flow measurement inaccuracies due to fouling of venturis. The plant-specific Byron and Braidwood AMAG (Advance Measurement and Analyses Group Inc.) cross flow Instrumentation was installed in accordance with the vendor specified installation requirements. The plant specific AMAG equipment accuracy results are documented in Westinghouse calculations CN-PS-03-29 Rev. 0 for Byron Unit 1 and CN-PS-03-30 Rev. 0 for Byron Unit 2 and Westinghouse calculations CN-PS-03-18 Rev. 1 for Braidwood Unit 1 and CN-PS-03-31 Rev. 0 for Braidwood Unit 2 and Byron/Braidwood calculation NED-I-EIC-0233 Rev 001A.

50.59 REVIEW COVERSHEET FORM

Station: Byron/Braidwood Units 1&2

Activity/Document Number: AMAG Implementation Using FW Header Flow **Revision Number:** 3

Title: Safety Evaluation for the Implementation of the AMAG Correction Factor

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

New procedure BVP 800-47 for Byron, and BwVP 850-26 for Braidwood, will be used periodically to obtain ultrasonic feedwater flow measurements at the feedwater header. Several other utilities have used ultrasonic flow measurement systems mounted on the feedwater header to more accurately measure feedwater flow and develop correction factors. Correction factors are developed by taking the ratio of (ultrasonic common header flow minus the sum of tempering flows) by the (sum of raw venturi flows on each feedwater line), as specified in procedures BwVP 850-26 (Braidwood) and BVP 800-47 (Byron). The Operations department is responsible for manually entering the calculated correction factors into the plant process computer to correct for feedwater flow inaccuracies. These same correction factors are used in the daily calorimetric calculations. The correction factor will be used in the plant process computer and daily calorimetric calculations, and will remain valid until a new correction factor is determined by subsequent ultrasonic feedwater flow measurements, or when the Operations Shift Supervisor determines that the use of the correction factor is not appropriate based on plant operating conditions. The following procedures and Operator Aids are being issued and/or revised for the use of AMAG system on common feedwater header.

1BOSR 3.1.2-1	Calorimetric Calculation Daily Surveillance
2BOSR 3.1.2-1	Calorimetric Calculation Daily Surveillance
BVP 800-47	Feedwater Venturi Calibration with Feedwater Header(new Procedure)
BOP FW-25	Feedwater Flow Correction factor
1(2)BwOSR 3.3.1.2-1	Power Range High Flux Setpoint Daily Channel Calibration (Computer)
1(2)BwOSR 3.3.1.2-2	Power Range High Flux Setpoint Daily Channel Calibration (Hand)
BwVP 850-26	Feedwater Venturi Calibration with Feedwater Header(New Procedure)
BwOP FW-26	Changing Main Feedwater Flow Calibration Multipliers (New Procedure)
OP Aids 99-044 and 99-045	Feedwater Flow Correction factor

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

The proposed activity is undertaken to eliminate noise contamination and correct overly conservative reactor thermal power calculations that result from the uncorrected feedwater flow venturi readings that are biased because of fouling and other venturi induced measurement inaccuracies. The AMAG ultrasonic flow measurement system is used as a "calibration" tool to improve the accuracy of the venturis at the plant process computer in order to obtain more accurate feedwater flow readings. The ultrasonic flow measuring devices are not affected by fouling and other venturi induced measurement inaccuracies. Therefore, these measurements can be used to correct the venturi readings (application of correction factors), and to obtain a more accurate calorimetric reactor thermal power calculation. This helps the station to operate closer to the licensed calorimetric rating. The method of data collection and application described herein for the AMAG ultrasonic instrumentation located on the feedwater header, is consistent with the current method of data collection and application for the AMAG ultrasonic instrumentation located on the individual feedwater lines.

Due to signal contamination, the ultrasonic sensors installed on the individual feedwater lines upstream of venturis do not provide a quality measurement. However, AMAG has confirmed that no noise contamination is present on the feedwater header installation (Reference: AMAG Letter dated 9/16/03). Therefore, the AMAG ultrasonic transducers have been relocated to the feedwater common header.

50.59 REVIEW COVERSHEET FORM

Station: Byron/Braidwood Units 1&2

Activity/Document Number: AMAG Implementation Using FW Header Flow **Revision Number:** 3

Title: Safety Evaluation for the Implementation of the AMAG Correction Factor

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

The Plant Computer accepts input from various plant instruments, including flow information from the feedwater venturis, and uses this information to determine the calorimetric power level of the reactor. Calorimetric calculations are performed routinely to determine reactor power.

The use of the ultrasonic feedwater flow measuring equipment, in-and-of-itself, has no affect on nuclear safety. The ultrasonic data acquistion/analyzing components are connected to the permanently installed transducers, and are used to obtain feedwater flow measurements. Whenever this data is used to correct the venturi feedwater flow in the calorimetric calculations as a function of actual reactor thermal power level, the appropriate procedural and administrative controls are utilized to ensure there are no adverse effects on nuclear safety.
As stated previously, the use of ultrasonic flow measurements, which are not subject to venturi fouling and other venturi induced measurement uncertainties, provide an effective way to calibrate the feedwater venturi flows used in the calorimetric calculation of "actual reactor thermal power" to obtain recovery of lost generating capacity while staying with plant licensed limits. The ultrasonic correction factor allows the plant to operate closer to 100% rated thermal power. The development and use of ultrasonic correction factors will be adminisratively controlled to ensure that the reactor will not operate at levels higher than rated 100% thermal power as indicated by the calorimetric power.

The Model for Flow Accelerated Corrosion for secondary side is based on maximum design flow conditions. The use of AMAG will not exeed this limit.

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

Calculation NED-I-EIC-0233, Daily power Calorimetric Accuracy Calculation Rev. 001A, evaluated the impact of using the AMAG ultrasonic flow instrumentation on the 2% RTP error margin, and concluded that the use of the AMAG instrumentation would not increase the total error uncertainty above the 2% error margin. Therefore, since the error does not exceed the analyzed 2% margin, the existing core analysis is satisfied.

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

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

The Byron/Braidwood spent fuel criticality analysis includes 5% uncertainty in the calculated assembly burnups. This is conservative with respect to the 2% reactor power measurement uncertainty, and as assembly burnup is determined as an integral of reactor power, the 2% uncertainty would bound the reactor power being overstated

Station: Byron/Braidwood Units 1&2

Activity/Document Number: AMAG Implementation Using FW Header Flow **Revision Number:** 3

Title: Safety Evaluation for the Implementation of the AMAG Correction Factor

throughout the fuels operating history. As long as the reactor power and associated integral fuel burnup are established with a method that satisfies the 2% assumed measurement uncertainty, the criticality analysis is satisfied.

If defouling of the venturi occurs, the guidance provided in Operator Aid and Operating Procedure BOP FW-25 for Byron and BwOP FW-26 for Braidwood, will direct the reactor operator to nullifying the effect of the ultrasonic flow measurements and request that a new set of ultrasonic flow measurements be taken. Additionally, based upon plant operating parameters, the reactor operator may at any time elect to nullify the correction factor to ensure the conservative and reliable operation of the plant.

The ultrasonic flow measurement data is evaluated periodically and the feedwater flow correction factor is applicable to Mode 1 of plant operation. While the factor is present in the plant computer and is used in the daily calorimetric calculation, it has proportionately less impact at lower power levels. An average correction factor greater than 1.00 is unexpected and will be investigated before any further corrections are applied.

Summary of Conclusion for the Activity's 50.59 Review:

The Proposed Activity does not require NRC notification prior to implementation. Use of the flow data obtained from the AMAG feedwater ultrasonic flow equipment when operating as intended and used in accordance with the established procedures and operator aids, will not exceed the operating limit of 100% RTP. Therefore, the proposed activity will not result in an increase in frequency of occurrence of an accidents previously evaluated in the UFSAR, will not result in an increase in the likelihood of occurrence of a malfunction, will not increase the consequences of a malfunction, will not result in the possibility of a malfunction with a different result of an SSC important to safety previously evaluated, or create new failure modes. Since the basis for accident/transient analyses contained in the UFSAR are maintained and remains valid, and no new accidents are created, the proposed activity does not result in exceeding or altering the Design Basis Limit for the Fission Product Barrier (DBLFPB). The proposed activity does not result in a departure from the method of evaluation described in the UFSAR used in establishing the design basis or in the accident analyses because method of calculating uncertainties associated with reactor power was not changed.

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

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
X	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-03-0007 (Byron) BRW-E-2003-218 (Braidwood)	**Rev.**	3

BRW-E-2003-220

50.59 REVIEW COVERSHEET FORM

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

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

Revision Number: 0

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

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

*

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

*

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

*

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

*

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

*

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

*

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

*

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

*

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

*

There is no effect on plant operations.

*

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

*

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

*

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

*

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

*

50.59 REVIEW COVERSHEET FORM

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

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

Revision Number: 0

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

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

*

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

*

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

*

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

*

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

*

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

*

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

Forms Attached: (Check all that apply.)

☐	Applicability Review				
☐	50.59 Screening	50.59 Screening No.		Rev.	
✓	50.59 Evaluation	50.59 Evaluation No.	BRW-E-2003-220 / 6G-03-008	Rev.	0

BRW-E-2003-229

50.59 REVIEW COVERSHEET FORM

Station: Braidwood Unit 1

Activity /Document Number: TRM Change #03-015 **Revision Number:** N/A

Title: Change In-Core Decay Time for A2R10

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 A2R10 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 A1R10)." This activity will revise this statement by replacing "(≥ 65 hours for A1R10)" with "(≥ 65 hours for A2R10)."
- Condition A under TRM 3.9.a states "Reactor subcritical for < 100 hours (< 65 hours for A1R10)". This activity will replace "(< 65 hours for A1R10)" with "(< 65 hours for A2R10)".
- Surveillance requirement TSR 3.9.a.1 will be revised by replacing "≥ 65 hours for A1R10" with "≥ 65 hours for A2R10".

Reason for Activity:

Discuss why the proposed activity is being performed)

It is anticipated that during A2R10, 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 A2R10.

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. 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 A2R10 still meets this constraint.

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 A2R10 Fuel offloading activities earlier than 100 hours.

50.59 REVIEW COVERSHEET FORM

Activity /Document Number: TRM Change #03-015 **Revision Number:** N/A

Title: Change In-Core Decay Time for A2R10

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 A2R10.

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 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

50.59 REVIEW COVERSHEET FORM

Activity /Document Number: TRM Change #03-015 **Revision Number:** N/A

Title: Change In-Core Decay Time for A2R10

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 A2R10 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:

'tach all 50.59 Review forms completed, as appropriate.
).ote: 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-2003-229	Rev.	0