DOCUMENT CHANGE REQUEST

990849

DESCRIPTION

The purpose of this Document Change Request No. 990849 is processed to correct a typographical error in P&ID M-48 sheet 6B, which is UFSAR Figure 11.2-7 sheet 2. Specifically the EPN for Valve 1RF060 has been designated as 2RF060.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because Document Change Request No 990849 is processd to correct a typographical error on a P&ID that is also a UFSAR figure. Therefore, in accordance with procedure LS-AA-104 a Safety Evaluation is required There is no impact on any UFSAR accidents or their consequences. In addition, there is no impact on any equipment or the consequences of equipment malfunction.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because Document Change Request No 990849 is processed to correct a typographical error on a P&ID that is also a UFSAR figure. Therefore, the possibility of a different type of an accident or malfunction than previously evaluated is not created

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because there is no impact on any Technical Specifications Document Change Request No. 990849 is processed to correct a typographical error on a P&ID that is also a UFSAR figure.

DOCUMENT CHANGE REQUEST

990846

DESCRIPTION

The purpose of this DCR is to incorporate changes evaluated under ER9903221 and installed under WR#990199169. This change relocated the battery pack for emergency light 1LL1-112 to the opposite side of col. G-2 to provide safer access for the required maintenance activities. This change has been performed as a Non-power Block change and has been evaluated under 10CFR50.59 Screening No. BRW-FCS-2000-782. This change does not affect the function of the Local Lighting (LL) system or any other SSC relied upon to mitigate an accident or transient. This change has no affect on nuclear safety.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the subject emergency light battery pack does not impact the function of the LL system. The LL system is not related to the sequence of events leading to the initiation of any of the analyzed accidents. Therefore, the proposed DCR does not increase the probability of occurrence of any accident or transient described in the UFSAR.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created since the subject items do not impact the intended function of the LL system. No adverse SSC interactions are introduced as a result of theses changes. Therefore, the possibility of an accident or a transient of a different type from those evaluated in the UFSAR is not created.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because all Technical Specification requirements were met.

Safety Evaluation Summary Form

Tracking No. **6G-01-0010**
Activity No. DRP 9-041

DESCRIPTION:

The following UFSAR changes were made:

- *Revised UFSAR Section 9.5.2.1 to add a description of the Emergency Notification System (ENS) and the Emergency Response Data System (ERDS).*
- *Revised UFSAR Section 9.2.1.2.1 to add a description of the Essential Service Water System opposite unit cross-tie.*
- *Revised UFSAR Section 10.4.7.5 to add a description of the Erosion/Corrosion inspection program.*
- *Revised UFSAR Section 6.3.2 to add valves 1/2SI8802A/B to list of valves evaluated for pressure locking.*

SAFETY EVALUATION SUMMARY:

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

   The UFSAR changes did not involve physical changes to the facility and did not physically alter the performance characteristics of any system or challenge any system in any manner that would change the initiators of any accident or transient described in the UFSAR. The UFSAR changes did not physically alter the performance characteristics or challenge the integrity of any system in any manner that would affect the integrity of any fission product barriers or any primary or secondary pressure retaining systems. These changes did not alter the capability of any system to perform its design function in the event of a design basis accident and did not affect any mitigating function assumed in the UFSAR. Therefore, calculated doses for design basis accidents were unaffected by the change and as a result, the UFSAR changes did not involve any increase in the consequences of any accident or transient. No new or different equipment was installed and no installed equipment is operated in a different manner.

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

   The UFSAR changes did not add new or different equipment to the facility. There were no changes to the methods utilized to respond to plant transients and no alterations to the way the plant is normally operated. The UFSAR changes did not alter instrumentation setpoints that initiate protective or mitigating actions. As a result, no new failure modes were introduced. The design, hardware, and performance characteristics of the various systems were unchanged by this activity

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

   The proposed UFSAR changes are consistent with the basis for Technical Specifications 3.5.2 and 3.7.8, thus the margin of safety was unchanged.

PEPP-E FORM

ENGINEERING REQUEST

9903911

DESCRIPTION

The purpose of this Engineering Request was to install a freeze seal on line 1SX58AA02" to install Design Change D20-1-00-330. This is the SX supply to the Unit 1A CV pump gear and lube oil coolers.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR is not increased, because the freeze seal has the same affect on the plant equipment as closing the SX supply isolation valves to coolers 1CV02SA & 1V03SA. The additional weight and flooding were evaluated and are not a concern, and there is no effect on overall SX system flow. The SX system is also not a radiological barrier. The work will be performed when the Unit 1A CV pump is Out-Of-Service.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the freeze seal has the same affect on the plant equipment as closing the SX supply isolation valves to coolers 1CV02SA & 1CV03SA. The additional weight and flooding were evaluated and are not a concern, and there is not effect on overall SX system flow. The work will be performed when the Unit 1A CV pump is Out-Of-Service.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the freeze seal has the same affect on the plant equipment as closing the SX supply isolation valves to coolers 1CV02SA & 1CV03SA. There is no impact on the SX system that would reduce its margin of safety from the installation of these freeze seals. The Unit 1 ECCS "B" train systems shall be operable to meet Technical Specification requirements should the freeze seal fail and cause the Unit 1 EECS "A" train coolers to be isolated from SX flow.

NUCLEAR WORK REQUEST

990197010

DESCRIPTION

The purpose of this Nuclear Work Request was to meet the requirements necessary for the removal of the 1SI01PB pump motor. To support this activity, doors D-269/280 will need to be propped open. These doors are HVAC boundary doors, thus will be evaluated per the plant barrier impairment procedure, CC-AA-201.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR is not increased, because the probability of an occurrence, or the consequences of an accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR, is not increased because propping open door D-269/D-280 does not have any impact on the events which initiate a Loss Of Coolant Accident (LOCA). This evaluation for propping open door D-269/D-280 ensures any design base accident remains bounded by the existing off-site dose boundary analysis calculation .

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the affected structure provides a ventilation boundary for Auxiliary Building HVAC (VA) systems. Part of the design requirements for the VA system is to limit environmental conditions in various zones in conformance with requirements The system controls radioactivity in the areas served by staging the supply air from the clean areas to the areas of greater potential contamination. Plant operation is not changed No new failure modes are introduced.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the change is administratively controlled upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety

ENGINEERING REQUEST

9903910

DESCRIPTION

The purpose of this Engineering Request was to install a freeze seal on the discharge side of Relief Valve 1CC9421B on line #1CC48A-2" to perform maintenance/replace the valve. The Unit 1A Letdown Heat Exchanger will be OOS.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR is not increased, because the installation of the freeze seal on this CC piping does not change any initiating condition or impact any accidents and transients evaluated in the UFSAR since the Unit 1A Letdown Heat Exchanger will be OOS. The added weight of the freeze seal has been evaluated along with flooding concerns and found acceptable. The CC system is not normally a radiological system but could possibly become contaminated from a Heat Exchanger tube rupture. Any leakage from a failed freeze would be contained in the Auxiliary Building and no increase in offsite dose would occur.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the temporary freeze seal installation does not impact any other plant equipment that could initiate or create an accident different from those evaluated in the UFSAR. The freeze seal does not affect required plant equipment since the overpressure protection function of the relief valve is not required with the 1A Letdown Heat Exchanger isolated and OOS.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the additional weight, flooding, and the effects a failed freeze seal on the CC system have all been previously addressed. The 1A Letdown Heat Exchanger will also be OOS during this time. The freeze seal will have an insignificant effect on CC system flow, even with a failed freeze.

ENGINEERING REQUEST

9903918

DESCRIPTION

The purpose of this Engineering Request was to revise Braidwood P&ID drawing M-49 Sheet 3 Revision W to properly depict the as built flowpath from the Makeup Demineralizer (WM) system carbon filters to the filtered water storage tank and the as built pipe diameter of sand filter drain pot overflow line 0WM81A

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR is not increased, because the makeup demineralizer has no safety functions No physical changes will be made to the field piping configuration or any plant equipment. The proposed activity is limited to drawing changes only. The drawing changes are consistent with operation of makeup demineralizer system to provide makeup to plant components as described in UFSAR Section 9.2.3. Therefore, failure modes of equipment important to safety are not increased. Accident probabilities or consequences are not affected by these changes.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the proposed activity is limited to drawing changes only. No physical changes will be made to the field piping configuration or any plant equipment The drawing changes are consistent with operation of makeup demineralizer system to provide makeup to plant components as described in UFSAR Section 9.2 3. Therefore, the possibility of a different type of malfunction is not created.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because no technical specification basis was identified as being affected.

DOCUMENT CHANGE REQUEST

990868

DESCRIPTION

The purpose of this Document Change Request #990868 is to revise drawings #M-195 Revision AF, S-1065 Revision N, and S-1070 Revision R to clarify the size of the screens for the Emergency Core Cooling System (ECCS) recirculation sumps, inside the Unit 1 and Unit 2 Containment buildings.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the probability of any accident or equipment malfunction is not increased since this change has no effect on any of the initiating factors for any accidents The consequences are not increased since the design function of the ECCS is maintained

   The different screen type (3/16 inch actual opening size) results in an open area through the inner screens of 49.1% The affected calculations have been reviewed. The open area through the inner screen that has been evaluated in DCR Type calculations #990628 and 990629 is 33%. The existing calculations are thus conservative and do not need to be revised.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because there is no change to any components/structures that would create the possibility of a different type of malfunction or accident.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the margin of safety is not affected DCR 990868 and DRP 9-027 provided updates to owner controlled documents to reflect the actual size, based on field measurements, for the outer and inner screens for the ECCS sumps inside Containment. A bounding size for the screens had already been evaluate under Safety Evaluation #BRW-SE-2000-1362 and Validation #BRW-SESV-2000-527. The evaluated bounding size has been verified to be applicable to all Byron and Braidwood Units.

TECHNICAL REQUIREMENTS MANUAL

01-001

DESCRIPTION

The purpose of this is to revise U1 and U2 Core Operating Limits Reports (COLRs), reference TRM Change No. 01-001.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the proposed activity changes the COLR DNB limit for Pressurizer pressure to use a value consistent with Chapter 15 analyses initial conditions In DNB analyses, a nominal pressure value of 2250 psia (2235 psig) is used. The uncertainties associated with power, pressure, temperature, and flow are considered using the Revised Thermal Design Procedure (RTDP), WCAP 11397-P-A. The uncertainty assumed for Pressurizer pressure in the analyses is 43 psi (reference NFM NDIT NFM9900208) for a low limit of 2207 psia (2192 psig) The proposed change is consistent with the assumptions provided by NFM TODI NFM0000188, "Pressurizer Pressure DNB Limit", which includes Pressurizer pressure indication uncertainty of 17 psig to derive a new Pressurizer Pressure DNB limit value of 2209 psig. Since the initial conditions used in the analyses have not changed, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because as stated as above, the initial conditions used in the analyses have not changed. Therefore, the possibility of an accident of a different type has not been created.

   The proposed activity does not change any equipment, setpoints, or plant operating parameters Since the proposed activity does not change any equipment, setpoints, or plant operating parameters, the possibility of a different type of malfunction of equipment important to safety has not been created.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because as stated above, the proposed activity is consistent with Chapter 15 analyses. The initial conditions used in the analyses have not changed. The functions of systems, structures, and components (SSCs) are not required to be changed to implement the revised COLRs. Plant operating parameters and setpoints are also not required to be changed to implement the revised COLRs. Therefore, no reduction in the margin of safety (as described in the basis for any technical specification) will occur by this activity.

UFSAR REVISION

UFSAR Draft Revision Package 9-026

DESCRIPTION

The purpose of this UFSAR Revision was to add some components on the Emergency Doesel Generator (EDG) skid to UFSAR Table A1.26-1. The subject table lists the components that are no longer available as ASME Section III items. The procurement of these components is in compliance with Regulatory Guide 1 26 as stated in UFSAR Appendix A1

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because no equipment was affected. This change was administrative in nature to accurately reflect plant conditions in the UFSAR in compliance with Regulatory Guide 1.26

2 The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the affected equipment, its operation and maintenance, are unaffected by the proposed activity. This change was administrative in nature to accurately reflect plant conditions in the UFSAR in compliance with Regulatory Guide 1.26.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because there are no Technical Specifications affected.

ENGINEERING REQUEST

9903962

DESCRIPTION

The purpose of this Engineering Request is to perform a Furmanite leak seal injection repair to stop the packing leak on valve 2FW090A during Unit 2 Operation The Unit will be reduced in power to approximately 35% during the repair. Valve 2FW090A is the drain valve for Steam Generator 2A, located inside Missile Barrier, Unit 2 Containment

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the Furmanite injection will seal the leak and render the valve inoperable in its normally closed position. The system integrity will be maintained, which is this valves only function during plant operation. This valve is only used during outages to assist in draining the 2A Steam Generator.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the Furmanite injection of valve 2FW090A will have no affect on other plant equipment or operations. The valve will be inoperable in the closed position (normal position) and maintain system integrity Alternate means are available to drain the 2A SG.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because this repair has no affect on Technical Specifications The valve will be inoperable in the closed position (normal position) and maintain system integrity, which is its only function during normal operation.

SPECIAL PROCESS PROCEDURE

00-023

DESCRIPTION

The purpose of this Special Process Procedure governs the use of the Fuel Handling Building Crane during Heavy Load Operations required for the removal and replacement of the spent fuel racks in and around the Spent Fuel Pool. This Special Procedure also governs the adjustments to the Main and Aux Hooks height, in order to be able to lift the old and replacement racks high enough to clear obstructions on the 426 elevation of the fuel handling building

Normally, BwFP FH-20T1 "Fuel Handling Operating Zones for Heavy Loads", describes the allowable heavy load paths in the Fuel Handling Building. However, for this SFP re-rack activities, special heavy loads paths have been created for the removal of the current racks and insertion of the new racks in accordance with NUREG 0612 guidance This procedure refers to the Holtec Procedure HPP-80944-40 "On-Site Handling, Installation and Removal Procedure" for the Actual load paths

Adjusting the height of the fuel handling building crane does not effect the normal operation of the Fuel Handling Crane or activities in the Fuel Building SPP 00-023 is a master overview Procedure. The following functions are included.

- Provides the directions for performing crane checkouts and operation per BwFP FH-20.
- Verifies that the work is planned, Documented, discussed and understood for heavy load paths
- Verifies heavy load path does not pass over fuel in the SFP
- Verifies that adequate decay time of the fuel in the pool is Greater than 90 days and that the fuel is positioned away from heavy load paths.
- Delineates the crane operation on the designated Heavy Load Paths, on the New Fuel Vault storage area and over the Spent Fuel Pool.
- Provides a log for documenting activities under this procedure
- Provides guidance for adjusting the height of the Fuel Handling Building Crane Main and Aux hook.
- Provides Acceptance Criteria with respect to No Heavy Loads being carried over fuel and crane restoration.

In addition, DRP 8-087 changed the UFSAR section 15.7.5.2 to allow the use of the unrestrained main hoof of the FH building crane provided it is not moved over fuel in the SFP. This change was necessary for this SPP.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because heavy loads are maintained within safe load paths by procedures. Safe load travel paths are reinforced during crane use by detailed briefings. The extended boundary for the safe load path in SPP 00-023 is acceptable since all fuel assemblies are removed from below

the safe load path. A special travel path for the FME barrier is shown in Holtec procedure HPP 80944-40, which SPP00-023 references to ensure the barrier will not fall into the SFP during rack removal and replacement The spent fuel in the pool is placed sufficiently apart from the safe load path and the fuel in the pool.

The height increase of the main and Aux hook will not increase the probability of an accident or malfunction of equipment important to safety. This is required in order to clear fixed obstructions surrounding the spent fuel pool area

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because there are no new of different events created by SPP 00-023 or by changes to the permanent plat BwFP FH-20 series of procedures. Heavy loads are maintained within safe load paths and heavy load drops within safe load paths have been previously evaluated in the SAR The altered safe load path for SPP 00-023 maintains the same or greater distance to fuel as shown on station drawings for the existing safe load path along the North side of the Spent fuel pool. The increase height of the Main and Aux hook for the fuel handling building crane does not effect any evaluated safety analysis due to the height of the analyzed drop or event was estimated at 30 feet. The height was required in order to clear obstructions surrounding the spent fuel pool area. The maximum height of the load from the floor to the bottom of the load will be minimized around the area of the spent fuel pool. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis is not created.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the SPP and the revised permanent plant procedures ensures loads greater than 200 lbs , including the unrestrained crane main hook, do not travel over fuel in the spent fuel pool. The height of the main and AUX hook is not effecting the operation of the crane not effecting any Technical Specifications. No Technical Specifications are affected, therefore the margin of safety is not reduced as defined in the basis.

PLANT BARRIER IMPAIRMENT

DESCRIPTION

The purpose of this Plant Barrier Impairment in connection with Nuclear Work Request 990238510-01 requires opening floor plug 2SXFS01-3 for maintenance activities, which will connect auxiliary building 346' general area with the 330' 1A/2A SX pump room These doors are considered HVAC boundary doors and will be evaluated in accordance with the plant barrier impairment program.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the accident mitigation function and normal functions of the VA system are unaffected since 1) the building will continue to be maintained at a negative pressure, 2) post accident radioactivity leaking from the ECCS equipment will be controlled within the required limits, and 3) EQ Zone requirements will be maintained for the affected areas.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the Auxiliary Building HVAC system and floor plug 2SXFSO1-3 are unrelated to the sequence of events leading to the initiation of an accident. Since the system's accident mitigation and normal functions will be maintained, the possibility of an accident or malfunction of a different type from those previously evaluated in the SAR is not created.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the change does not affect the Technical Specifications. There are no Technical Specifications for differential pressure or temperature limits for these areas

PROCEDURE REVISION

BwAP 1205-14

DESCRIPTION

The purpose of this Procedure Revision was to add corporate Emergency Planning (EP) procedures, which were previously exempted from further 50.59 review, to the list of exempted procedures contained in BwAP 1205-14. A previous activity exempted the subject corporate GSEP procedures from future 50.59 Safety Evaluations by performing a one-time evaluation. This activity provides for the addition of the exempted procedures to Station procedure BwAP 1205-14, which lists exempted activities and their associated OSR or ITR numbers. The listing of the exempted activities in the Station procedure is completely encompassed by the screening performed to exempt the listed corporate Emergency Planning (EP) procedures.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because this activity is administrative only.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because no new failure modes or conditions are created by the implementation of this activity.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because this activity does not affect any parameters upon which Technical Specifications are based.

Safety Evaluation Summary Form

Tracking No. BRW-SE-2001-59
Activity No. Technical Requirements Manual Change Request #00-018

DESCRIPTION:
Revise Technical Requirements Manual (TRM) Limiting Condition of Operation (TLCO) 3.1.k, "Position Indication System – Shutdown (Special Test Exception)," for clarification of requirements to ensure Condition A is not inappropriately entered.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because:
   This TRM revision provides clarification via the use of Condition Notes to ensure that Condition A of TRM Special Test Exception TLCO 3.1.k is not inappropriately entered. This change does not involve a physical alteration to the plant, nor does it affect the function of any system, structure, or component (SSC). Thus, the probability of occurrence of Digital Rod Position Indication failures is not affected. In Modes 1 and 2, the Position Indication System is used to detect a dropped assembly or bank or misaligned assemblies. However, in Modes 3, 4, and 5, the Position Indication System is not used for, nor is it capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design bases accident (DBA). The Position Indication System is not used to monitor a process variable, or the status of any design feature, or operating restriction that is an initial condition of a DBA or transient. The Position Indication System is not part of a primary success path in the mitigation of a DBA or transient.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because:
   This change does not involve a physical alteration to the plant. No new equipment is being introduced and no installed equipment is being operated in a new or different manner. Consequently, this change does not affect the function of any SSC and will not result in any changed interactions with other SSCs.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because:
   The Position Indication System requirements during Modes 3, 4 and 5, and the associated Special Test Exception were determined not to meet any of the criteria for Technical Specification inclusion as described in 10 CFR 50.36. Consequently, these requirements reside in the TRM. The TRM wording change provides clarification of TRM requirements and has no affect on any Technical Specification.

TECHNICAL SPECIFICATION BASES REVISION

(B 3.5.1)

DESCRIPTION

The purpose of this Technical Specification Bases Revision was to revise the Bases Section B 3.5.1, "Accumulators" to clarify the assumptions of the current LBLOCA and SBLOCA Analysis of Record (AOR)

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the Technical Specification Bases revisions proposed do not create any condition that would challenge the design or operational limitations of the affected equipment in the ECCS system The proposed activity will not increase the probability of occurrence of a malfunction of equipment important to safety. The ECCS system remains capable of performing its design accident mitigation function. No functional changes operational modes, or failure mechanisms are introduced by the proposed revision, therefore, the consequence of a malfunction of equipment important to safety will not be increased.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the proposed Technical Specification Bases revision does not change the function of the ECCS system and affected components. There are no direct or indirect impacts on interfacing systems and components Since no new failure modes are created by the changes, there will be no malfunction of equipment of a type different from those evaluated in the UFSAR

3 The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the ECCS system calculated cooling performance must meet the acceptance criteria as specified in 10CFR50.46(b) Approved modeling methods must include supporting justification to show that the analytical technique utilized realistically demonstrate the behavior of the RCS and the ECCS systems during a postulated accident Comparisons are made to applicable experimental data and uncertainties in the analysis method and inputs Once identified, results are compared to the ECCS acceptance criteria with a high level of probability that the limits will not be exceeded. If it is determined that inputs (assumptions) were not incorporated, or may have resulted in error, corrections are performed and the new results are compared to the 10CFR50.46(b) Acceptance Criteria. Errors are not considered to be significant if the resulting Peak Cladding Temperature (PCT) does not change collectively or individually by more than 50°F. These errors are reported to the NRC annually.

   The PCT penalty for the water injection delay as described by this proposed activity has resulted in a 20°F penalty for Unit 2 only. This penalty combined with previously identified input parameter errors have increased the total PCT error to 46°F, and thus a new Unit 2 SBLOCA PCT value of 1871°F. The resultant PCT change remains "insignificant" and well below the acceptance criteria limit of 2200°F as specified in 10CFR50.46(b). Therefore, the Margin of Safety is NOT reduced.

TEMPORARY MODIFICATION

00-0-005

DESCRIPTION

The purpose of this Temporary Modification (TMOD) was to provide temporary cooling to the Chemistry Counting Room, low level laboratory and the offices during maintenance activities to the Laboratory Ventilation (VL). The temporary cooling units will be powered from non-ESF lighting cabinets 112A & 112B. Door D-356 will be impaired during the period in which the VL system is Out-of-Service.

TMOD 00-0-005 is intended to maintain the environmental conditions within the counting room to support analysis of radiochemistry samples required by Technical Specifications and provide partial cooling for the low level lab and office areas. These installations do not impact any Technical Specifications.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the subject uninstalled filter does not impact the function of the LL system. The LL system is not related to the sequence of events leading to the initiation of any of the analyzed accidents. Therefore, the proposed DCR does not increase the probability of occurrence of any accident or transient described in the UFSAR.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created since the subject items do not impact the intended function of the LL system. No adverse SSC interactions are introduced as a result of theses changes. Therefore, the possibility of an accident or a transient of a different type from those evaluated in the UFSAR is not created.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because all Technical Specification requirements were met.

PLANT BARRIER IMPAIRMENT

DESCRIPTION

The purpose of this Plant Barrier Impairment in connection with Nuclear Work Request 990020347-02 requires opening doors D-306 & D-316 to allow for routing of hoses from the auxiliary building 401' general area through the U2 containment chiller room, into the radwaste HVAC equipment room for maintenance activities. These doors are considered HVAC boundary doors and will be evaluated in accordance with the plant barrier impairment program.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the accident mitigation function and normal functions of the VA system are unaffected since 1) the building will continue to be maintained at a negative pressure, 2) post accident radioactivity leaking from the ECCS equipment will be controlled within the required limits, and 3) EQ Zone requirements will be maintained for the affected areas

2 The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the Auxiliary Building HVAC system and doors D-306 & D-316 are unrelated to the sequence of events leading to the initiation of an accident Since the system's accident mitigation and normal functions will be maintained, the possibility of an accident or malfunction of a different type from those previously evaluated in the SAR is not created.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the change does not affect the Technical Specifications There are no Technical Specifications for differential pressure or temperature limits for these areas.

DESIGN CHANGE

980360

DESCRIPTION

The purpose of this Design Change Request was to revise drawings M-35 sheet 3 (Unit 1) and M-120, sheet 3 (Unit 2) to indicate the size of lines 1/2MS144A, 1/2MS80AA, 1/2MS80BA, and 1/2MS80CA as ½" diameter. The same change will be made electronically in EWCS. EWCS will also be revised to show the correct size of valves for 1/2MS168, 1/2MS098A, 1/2MS098B, and 1/2MS098C. Drawings M-2035 sheet 9 and M-2120 sheet 9 will also be revised to remove the pipe reducer located on the dowstream side of the instrument root isolation valves Additionally, drawings M-35 sheet 3, M-120 sheet 3, M-2035 sheet 9, and M-2120 sheet 9, will be revised to show lines 1/2MS144A connecting to the Hot Reheat piping downstream of the intercept valve.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the proposed activity will change existing drawings to reflect the as built plant configuration. All proposed changes under DCR 980360 are associated with instrumentation that has historically shown to function satisfactorily under the existing configuration No accident or anticipated transients have been identified to be affected by this proposed activity. Therefore, the changes being proposed by this DCR will not increase the probability of occurrence of any accident or transient as detailed in the UFSAR.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created no accidents or transients were identified based on the changes proposed by this DCR. Therefore, the proposed activity will not create the possibility of an accident or transient of a different type than previously evaluated.

   The proposed activity will correct existing station drawings to reflect the current plant configuration. No new failure modes will be created through the changes made by this DCR. Historical performance of equipment affected by this DCR has not indicated an increased failure rate or incidence or malfunction.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because revising the associated drawings for the referenced instrument sensing lines and valves from ¾" to ½" does not change the function of the system. Historical operating data has shown that the as built configuration functions as expected. Additionally, the associated components affected by this document change are not used in the basis of the Technical Specification.

DOCUMENT CHANGE NOTICE

DCN 00131M

DESCRIPTION

The purpose of this Document Change Notice was to address the addition of the following Holtec International procedures to Exelon Work Packages for the replacement of spend fuel pool storage racks at Braidwood....

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the above listed procedures implement the installation of DCN 00131M. The probability of various accidents and occurrences during the installation of new spent fuel racks at Braidwood were considered during the development of DCN 00131M and its associated safety evaluation. The procedures listed above in conjunction with existing Exelon procedures ensure that the assumptions are requirements (including NUREG-0612) for DCN 00131M are maintained during the replacement process. Therefore, the probability or consequences of an accident are not increased by the use of these procedures.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created the above listed procedures implement the installation of DCN 00131M. The possibilities for an accident or malfunction of a different type than any previously in the UFSAR during the installation of new spent fuel racks at Braidwood were considered during the development of DCN 00131 and its associated safety evaluation. The procedures listed above in conjunction with existing Exelon procedures ensure that the assumptions and requirements (including NUREG-0612) possibilities for an accident or malfunction of a different type than any previously in the UFSAR during the installation of new spent fuel racks at Braidwood are not changed by the use of these procedures.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the above listed procedures implement the installation of DCN 00131M. The possible reduction in the margin of safety during the installation of new spent fuel racks at Braidwood were considered during the development of DCN 00131M and its associated safety evaluation. As a result of this DCN, revisions to the Technical Specification were required as documented in Amendment #105 for both Braidwood Units 1 and 2. The procedures listed above in conjunction with existing Exelon procedures ensure that the assumptions and requirements (including NUREG-0612) for DCN 00131M are maintained during the replacement process. Therefore, the reduction in the margin of safety during the installation of new spent fuel racks at Braidwood is not changed by the use of these procedures.

Tracking Number: BRW-SESV-2001-81

PROCEDURE REVISION

BwAR 1-12-C1, Rev 6
BwAR 2-12-C1, Rev 6

DESCRIPTION

The purpose of this Procedure Revision was to update procedure setpoints (from 9.4% to 15.6%) for the Master controller Output that, controls when all Pressurizer Back-up Heaters are to be on. This change incorporates setpoints from approved SSCRs (SSCR #00-124 & SSCR #00-1 25).

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because a safety evaluation and supporting analyses (both Current operating cycle @ 3411 MWt and Uprate cycles @ 3586.6 MWt) have shown that the proposed changes to the Pressurizer backup heaters setpoints will have no adverse impact on the safety analyses. The results and conclusions presented in the UFSAR remain valid and bounding, and all applicable design and safety-limits continue to be met. Analyses by NFM has shown that changing the backup heaters "on" setpoint such that the heaters turn "on" sooner will have negligible impact on margin to hot leg saturation. The current analysis AND uprated analysis is conservative and the Safety criteria continues to be met. Therefore. the proposed activity does not increase the consequences of any accident or transient.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because changing the backuo heaters "on" setpoint will not increase the possibility of an accident or transient of a different one than prewously evaluated. The potential consequences of the proposed changes to the Pressurizer pressure control system setpoints are bounded by the current analyses AND uprated analyses.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the analyses by NFM has shown that changing the backup heaters "on" setpoint such that the heaters turn on sooner will have negligible impact on margin to hot leg saturation. The current analysis AND uprated analysis is conservative and the safety criteria continues to be met. No Technical Specifications or TRM sections are affected by implementing this activity.

ENGINEERING REQUEST

9903904
9903394

DESCRIPTION

The purpose of this activity is to extend the working length of the manufacturer's hoist pendant control pushbutton cable, from 20 feet to a maximum of 70 feet, for General Maintenance Crane SG #12(13), 1(2)HC22G per ER9903904, WR#990243220 (1HC22G) and ER9903394 WR#990175834 (2HC22G). This will be accomplished by the replacement of the existing cable with a longer cable. DCR990862 incorporates the new installed cable length for 2HC22G and provides an approved alternate length for 1HC22G.

Increasing the length of the hoist pendant control pushbutton cables does not alter the function of the general maintenance crane or have any adverse affect on any other SSC. This change only provides a greater area for operator positioning when the crane is in use. The additional length of cable has been evaluated per NEP-04-07, attachment 1, and does not affect the combustible loading analysis for this area. The cable will be coiled and stored per seismic housekeeping procedures when the crane is not in use.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the subject hoist does not impact the function of any system. The hoist is not related to the sequence of events leading to the initiation of any of the analyzed accidents. Therefore, the proposed activity does not increase the probability of occurrence of any accident or transient described in the UFSAR.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created since the subject items do not impact the intended function of the system. No adverse SSC interactions are introduced as a result of theses changes. Therefore, the possibility of an accident or a transient of a different type from those evaluated in the UFSAR is not created.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because all Technical Specification requirements were met.

Tracking Number: BRW-SESV-2001-91

DESIGN CHANGE

D20-2-01-011

DESCRIPTION

The purpose of this Design Change is to add a crosstie between the normal hydrogen supply and the nitrogen supply piping to the Volume Control Tank (VCT). The crosstie connects hydrogen supply line (2HY06A-1") upstream of the hydrogen supply pressure control valve (2CV8156) to the nitrogen supply line (2NT28A-1") upstream of the nitrogen supply pressure control valve (2CV8155). Pressure control valve 2CV8155 will maintain hydrogen cover pressure in the VCT at approximately 17 lbs. To support this, the controller for valve 2CV8155 to be recalibrated to match the setup for the 2CV8156 controller.

In order to install the crosstie, the removable spool piece upstream of valve 2CV8156 will be replaced with a modified spool piece consisting of a blind flange, threaded flange, and a tee that will connect to a threaded flex hose. Two diaphragm valves are added, one on the crosstie line and one on the nitrogen supply line upstream of the crosstie connection to ensure adequate isolation capabilities. Flexible hoses will be utilized to connect the hydrogen line to the nitrogen line.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the design change configuration will operate the same as that for the normal configuration.. Only the flow path is different, utilizing the nitrogen pressure control valve to regulate hydrogen supply pressure to the VCT. This change does not alter the initial conditions of any accident, does not change an SSC that is assumed to function during or after an accident, and does not change an SSC whose failure could lead to an accident.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the design change configuration will operate the same as that for the normal configuration, but through an alternate flow path. Isolation valves and administrative controls will ensure that the hydrogen does not contaminate the nitrogen system.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the installation of the design change does not affect any parameters upon which the Technical Specifications are based.

DESIGN CHANGE

9900658

DESCRIPTION

The purpose of this Design Change Package (DCP) was to rescale the Auxiliary Building ventilation system flow control loops 0F-VA711, 0F -VA712, 0F-VA722, and 0F-VA723, in support of modification DCP 9900472. The existing transmitters will be calibrated within their adjustable range. The transmitters supply a signal to the square root extractor which converts the linear velocity pressure signal into a comparable volumetric signal corresponding to a square root function. The volumetric signal output of the square root extractor is supplied to a controller and in three of the loops, to a flow indicator. To properly scale the flow control loops, three flow indicators and/or their scale indicator face plates, will be changed to provide a full range of flow indication starting at 0 cfm and ranging to the flow volume currently indicated at each flow indicator. The flow ranges indicated on the face of the flow indicators (exception is 0FI-VA723 which is currently at 0 to 10,000 cfm), currently begin at midrange which does not scale correctly because of the limitation on the square root extractor. This does not alter the intended function of the flow control loops and will provide the full range of flow indication consistent with system operation.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UIFSAR, is not increased because the VA System does no initiate or alter the initial conditions of any accident or transient. The VA fans are not required to run during a LOCA. The Charcoal Filter Booster fans are the fans credited during the accident. Therefore consequences are not increased.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the VA System is unrelated to the sequence of events leading to the initiation of any accident. Since the system's normal and accident functions will be maintained, the possibility of an accident or malfunction of a different type from those previously evaluated is not created. The airflow rate will maintain EQ zone temperature limits and airflow direction from clean to potentially contaminated areas.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the Non-Accessible Area Exhaust Filter Plenum Ventilation System and the Fuel Handling Exhaust Filter Plenum System are unaffected by this design change. However, a change is required to the testing flow rates for the filters (5.5.11). The new system airflows may be less than those required for testing. A Technical Specification Amendment may be submitted to change the flow test rates to conform to the ANSI N5.10-1980 testing criteria, if required.

DESIGN CHANGE PACKAGE

DCP 9900659

DESCRIPTION

The purpose of this Design Change Package was to lift leads for core exit thermocouples (CETCs) 9 (2TE-IT8001N) and 18 (2TE-8002B) in Train A Reactor Vessel Level Indication System/Core Exit Thermocouple (RVLIS/CETC) panel 2PA51J Lift heads for CETCs 40 (2TE-IT8001R) and 63 (2TE-IT8002Y) in Train B RVLIS/CETC panel 2PA52J. Disconnecting these inputa will ensure that data collection/trending of core power parameters and the calculation of reactor coolant system (RCS) subcooling margin are derived from valid thermocouple inputs.

SAFETY EVALUATION SUMMARY

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the SSCs affected by this change do not interact with other plant systems in a manner that could cause an accident or plant transient. Plant components whose failure could contribute to the initiation of the above postulated accidents are not influenced by this change. The CETCs do not provide an input to or control any of the equipment required to keep the offsite dose within 10CFR100 limits. Disconnecting the inputs from the subject CETCs does not adversely affect operation of equipment important to safety. This TMOD does not create any new interfaces with SSCs required to maintain the consequences within acceptable limits. There is no impact on release paths or offsite dose consequences as a result of this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the core exit thermocouple system will still be operational after implementation of this change. Four thermocouples that have shown erratic behavior are being removed as inputs to the subcooling margin monitor and plant process computer. Implementation of this TMOD will not cause any new failure modes or malfunctions beyond those that have already been considered. The proposed change will have no adverse effect on SSCs relied upon to mitigate the effects of postulated design basis events or transients No new failure modes have been identified for the CETCs or any other SSCs that they interface with.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because an evaluation was made of the minimum number of valid Core Exit Thermocouples (CETCs) necessary for measuring core cooling The evaluation determined the reduced complement of CETCs necessary to detect initial core recovery and trend the ensuing core heatup. The evaluation concluded that adequate core cooling is ensured by having four operational CETCs per quadrant. The minimum number of CETCs per quadrant will still be available after this change. Thus, the margin of safety is not reduced.

BASES REVISION

01-001

DESCRIPTION

The purpose of this Bases revision is to clarify information regarding Pressurizer pressure Departure from Nucleate Boiling (DNB) safety analysis assumptions and delete reference to the measurement uncertainty value.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR is not increased because the proposed activity clarifies the Pressurizer pressure DNB safety analysis assumptions consistent with Chapter 15 analyses initial conditions. In DNB analyses, a nominal pressure value of 2250 psia (2235 psig) is used. The uncertainties associated with power, pressure, temperature, and flow are considered using the Revised Thermal Design Procedure (RTDP), WCAP 11397-P-A. The uncertainty assumed for Pressurizer pressure in the analyses is 43 psi (reference NFM NDIT NFM9900208) for a low limit of 2207 psia (2192 psig). Since the initial conditions used in the analyses have not changed, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety.

2. The possibility of an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because as stated as above, the initial conditions used in the analyses have not changed. Therefore, the possibility of an accident of a different type has not been created.

   The proposed activity does not change any equipment, setpoints, or plant operating parameters. Since the proposed activity does not change any equipment, setpoints, or plant operating parameters, the possibility of a different type of malfunction of equipment important to safety has not been created.

3. The margin of safety, as defined in the Bases of the Technical Specification, is not reduced because as stated above, the proposed activity is consistent with Chapter 15 analyses. The initial conditions used in the analyses have not changed. The functions of systems, structures, and components (SSCs) are not required to be changed to implement the revised Bases revision. Plant operating parameters and setpoints are also not required to be changed to implement the revised Bases change. Therefore, no reduction in the margin of safety (as described in the basis for any technical specification) will occur by this activity.

WORK REQUEST

990232423
990231988

DESCRIPTION

The purpose of these Work Requests was to remove a floor plug for each tank at location elevation 401' in order to clean the 0WX01DA and 0WX01DB tanks. The floor plugs are considered part of the Auxiliary Building ventilation boundary and, therefore, removal of the plugs needs to be evaluated.

The normal VA design airflow path for the blowdown and radwaste mixed bed demineralizer rooms and valve operating areas (refer to P&ID M-95-5/UFSAR Figur 9.4-5 Sheet 5) is to draw supply air from the general area on elevation 383 into the radwaste and blowdown mixed bed demineralizer valve operating area (room #75). The air is then drawn into the radwaste and blowdown mixed bed demineralizer valve aisle (room #76), where the airflow path will be split between the radwaste mixed be demineralizer cubicles (room #77) and blowdown mixed bed demineralizer cubicles (room #78). Air from both these rooms is eventually exhausted to the VA non-accessible exhaust plenum

A bypass airflow path is created when the floor plug is removed to access either 0WX01DA and 0WX01DB, blowdown demineralizer tanks Air will be drawn from elevation 401 general area into room #78. This will cause a reduction in air drawn from elevation 383 general area into rooms #75-77 and exhausting the air into VA exhaust plenum. The lower airflows in these rooms has a potential to affect ALARA and temperature requirements

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because removal of the floor plug does not have any impact on the events which initiate a Loss of Coolant Accident (LOCA) Removal of the floor plug with the administrative controls in place ensures any design base accident remains bounded by the existing off-site dose boundary analysis calculation.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the affected structure provides a ventilation boundary for Auxiliary Building HVAC (VA) Systems. Part of the design requirements for the VA System is to limit environmental conditions in various zones in conformance with requirements. The system controls radioactivity in the areas served by staging the supply air from the clean areas to the areas of greater potential contamination. Plant operation is not changed. No new failure modes are introduced.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the change does not affect any parameters upon which Technical Specification are based; therefore, there is no reduction in the margin of safety.

PROCEDURE REVISION

BwOP RY-7
BwOP RY-11
1/2BwOSR 0.1-1,2,3

DESCRIPTION

The purpose of this Procedure Revision was to incorporate changes to U-1 and U-2 Core Operating Limits Reports (COLRs), reference TRM Change No. 01-001.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the proposed activity incorporates new COLR DNB limit for Pressurizer pressure into the Operating procedures to use a value consistent with Chapter 15 analyses initial conditions. In DNB analyses, a nominal pressure value of 2250 psia (2235 psig) is used. The uncertainties associated with power, pressure, temperature, and flow are considered using the Revised Thermal Design Procedure (RTDP), WCAP1139 7-P-A . The uncertainty assumed for Pressurizer pressure in the analyses is 43 psi (reference NFM NDIT NFM9900208) for a low limit of 2207psia (2192 psig). The proposed change is consistent with the assumptions provided by NFM TODI NFM0000188, "Pressurizer Pressure DNB Limit," which includes Pressurizer pressure indication uncertainty of 17psig to derive a new Pressurizer Pressure DNB limit value of 2209 psig. Since the initial conditions used in the analyses have not changed, there is no increase in the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the proposed activity does not change any equipment, setpoints, or plant operating parameters. Since the proposed activity does not change any equipment, setpoints, or plant operating parameters, the possibility of a different type of malfunction of equipment important to safety has not been created.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the proposed activity does not change any equipment, setpoints, or plant operating parameters. Since the proposed activity does not change any equipment, setpoints, or plant operating parameters, the possibility of a different type of malfunction of equipment important to safety has not been created..

WORK REQUEST

990060229

DESCRIPTION

The purpose of this Work Request was to remove the 2SXFS02-3 floor plug located on 346', N-19 in the Auxiliary Building. The floor plug is being removed to provide a path for hoses to support a freeze seal on the 1B SX cubicle cooler. The floor plug is considered an HVAC boundary for the 1B/2B SX pump room, therefore, will be evaluated under the plant barrier impairment program.

The normal mode of VA System design for this room involves two supply ducts providing supply air into the 1B/2B SX pump room and an exhaust duct providing and exhaust path back to the VA exhaust plenum where it is eventually discharded into the U-1 or U-2 stack. When the floor plug is removed, a bypass flow path will be crated from the general area 346' into the 1B/2B SX pump room. The VA System design requirements have been reviewed and determined acceptable to remove the floor plug. The basis for this validation is the Safety Evaluation, BRW-SE-1997-859, performed to evaluate HVAC effects to the VA System and the room when the system is operated under two fan configuration and extended Abnormal (booster fan only) operation.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased since the probability of creating an initiating event of LOOP or LOCA or HELB is not affected. Also, removing the floor plug will not affect any safety related equipment from a ventilation boundary perspective. The VA System will still meet its intended functions, thus all other safety related equipment will not be affected from ventilation concerns The airflow path essentially remains unchanged, thus VA will continue to function as before and the offsite dose analysis remains the bounding analysis.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created this action does not have an impact on the events which initiate a LOCA or a radioactive release accident

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the change does not affect any temperature or differential pressure requirements in the Technical Specifications

DESIGN CHANGE PACKAGE

DCP 9800070

DESCRIPTION

The purpose of this Design Change Package (DCP) 9800070 was to increase the design pressure of the discharge piping from each of the Diesel Fuel Oil (DO) Transfer Pump (1DO01PA/PB/PC/PD) from 50 psig to 60 psig This is needed to accommodate the expected backpressure of approximately 7.4 psig on the tailpipe of the relief valves in the discharge piping (Valve EPNs 1DO020A/B/C/D). Standing fuel in the relief valve tailpipe causes the backpressure. Note 7 will be added to drawing #M-50 sh 1A to indicate the actual lift pressure of relief valves 1DO020A/B/C/D. This note will also state the new design pressure of 60 psig.

Note 5 on drawing M-50 sh. 1A currently states a hydrotest pressure of 62.5 psig. This note will be modified to indicate that this requirement does not apply after the implementation of ASME Section XI code case N-498-1.

Evaluation #BRW-SESV-2001-157 is a validation of Safety Evaluation #BRW-SE-2000-529.

SAFETY EVALUATION SUMMARY

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the probability of any accident or equipment malfunction is not increased since this change has no effect on any of the initiating factors for any accidents. The consequences are not increased since the design function of the Diesel Fuel Oil (DO) system and the Emergency Diesel Generators is maintained.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the changes to the DO system implemented under DCP 9800070 do not have any adverse effect on the reliability of any interfacing system or supporting system. The changes do not introduce any new operational limitations for the affected diesel engine subsystem, nor do they challenge the availability of the Emergency Diesel Generators.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the implementation of DCP 9800070 does not challenge the reliability or availability of the Emergency Diesel Generators (EDGs) as a source of AC power. The reliability of the DO system piping integrity is not affected by this design change and the DO system remains capable of performing its function of maintaining an inventory of Diesel Fuel Oil in support of operation of the EDGs

Safety Evaluation Summary Form

Tracking No. BRW-SESV-2001-165
Activity No. C-Model Dose Assessment Program Product Version 3.1

DESCRIPTION:
Upgrades to the C-Model Dose Asessment Program include:
- All Zion modules deleted as they are no longer needed
- Modules for determination of release rate from the steam generator relief valves have been deleted as this capability is now provided by the B-model
- The module for calculation of radiation dose from an unplanned release of radioactivity to the aquatic environment has been deleted as this capability is not required by the GSEP
- The location database of pre-established monitoring locations has been updated to incorporate changes at LaSalle and Quad Cities
- The "other location" bug has been fixed. This bug sometimes caused a problem in back-calculation of noble gas release rate when the location used was not a pre-established monitoring location
- On-screen references to CEPIPs have been replaced by references to the Revised Standard Procedure EP-AA-110

SAFETY EVALUATION SUMMARY:

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

   The C-Model dose assessment software is not discussed in the SAR. The changes are to the software program only and do not affect plant equipment. Therefore, the probability of an occurrence or consequence of an accident or malfunction of equipment is not increased.

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

   The C-Model dose assessment software is not discussed in the SAR. The changes are to the software program only and do not affect plant equipment. Therefore, the possibility for an accident or malfunction of a different type is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because:
   The C-Model dose assessment program is not described in any of the basis of any Technical Specification.

Tracking Number: BRW-SESV-2001-166

PROCEDURE REVISION

BwOP RH-6

DESCRIPTION

The purpose of this Procedure Revision was to change the power restrictions for procedure performance. The limit was lowered to reach an equivalent thermal power output from the reactor prior to power uprate prior to byapssing the _7 heater.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because components will operate in a manner consistent with the current practice. The change to the procedure involve restrictions that are placed on the unit prior to allowing performance of the procedure. The value used (92%) was chosen to relate to a pre-power uprate secondary flow value equal to 97% power.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the change to the procedure is placing a new administrative limit on procedure performance. There is no change of intent or performance of the procedure, just the reactor power limit.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because this activity does not affect any parameters upon which the Technical Specifications are based.

WORK REQUEST

990159751

DESCRIPTION

The purpose of this Work Request was to work on the Unit 1 containment chilled water air seperator. Door D-341 will need to be propped open in order to route hoses and welding cables from the Auxiliary Building general area into the Unit 1 Spray Additive Tank room and Pipe Penetration areas. Door D-341 is considered part of the Auxiliary Building ventilation boundary and, therefore, propping opening of the door will need to be evaluated.

The normal VA design airflow path for the Unit 1 Spray Additive Tank room and Pipe Penetration area (refer to P&ID M-95, sheet 7/UFSAR Figure 9 4-5, sheet 7) is to supply air from the general area on elevation 346' into the Unit 1 Spray Additive Tank room and Pipe Penetration areas. The air is then drawn from the areas and eventually exhausted to the VA non-accessible exhaust plenum.

A bypass airflow path is created when door D-341 is opened Air will be drawn from elevation 401 general area into the Unit 1 Spray Additive Tank room and Pipe Penetration areas This will cause a reduction in air drawn from elevation 364'. The lower airflows in these rooms has a potential to affect ALARA, differential pressure, and temperature requirements.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because propping open door D-341 does not have any impact on the events which initiate a Loss of Coolant Accident (LOCA). This evaluation for propping open door D-341 ensures any design base accident remains bounded by the existing off-site dose boundary analysis calculation

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the affected structure provides a ventilation boundary for Auxiliary Building HVAC (VA) Systems. Part of the design requirements for the VA System is to limit environmental conditions in various zones in conformance with requirements. The system controls radioactivity in the areas served by staging the supply air from the clean areas to the areas of greater potential contamination. Plant operation is not changed No new failure modes are introduced.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the change is administratively controlled upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety.

DESIGN CHANGE PACKAGE/TEMPORARY MODIFICATION

9900666/01-1-001, 9900670/01-1-002, 9900667/01-2-002, 9900671/01-2-003

DESCRIPTION

The purpose of this Temporary Design Change is to implement changes to the control configuration for the FW EH Controller to support mini-uprate. Under this control configuration, the primary steam supply of the Turbine Driven Main Feedwater Pumps (TDMFPs), 1(2)FW01PB and 1(2)FW01PC is main steam and the backup (redundant) steam supply is not reheat. This change is accomplished by exchanging the positions of the LP Governor valve Servo Amplifier card C3 with the HP Governor valve Servo Amplifier card C4 in the FW Turbine EH Controller panels 1(2)PA36J and 1(2)PA37J, allowing for the calibration of the governor valve lift points to utilize main steam as the primary steam supply. This change will be restored to existing configuration during the subsequent refueling outage.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because in the loss of Normal Feedwater Flow Accident, a reduction of main feed flow is assumed as an accident precursor. The reduction in feed flow can be caused by equipment malfunction or loss of offsite AC power. The proposed activity has no affect on the loss of offsite power. Probability of equipment malfunctions related to the proposed activity are discussed under Questions 14 and 15 and impacts are concluded to be negligible. In the case of a loss of the TDMFWP(s) (the precursor potentially affected by the proposed activity), a reactor trip and subsequent steps in the accident analysis may be averted by a fast start of the motor driven MFWP or operator action to reduce load The probabilistic risk assessment (PRA) models as the accident initiator the failure of the motor driven feed pump to restore normal feedwater and assigns a probability of failure The proposed activity has no effect on the function of the motor driven feed pump, increase in the probability of occurrence o the Loss of Normal Feedwater accident and no other accidents or transients are affected.

   The probability of occurrence of malfunctions related to the use of the HP steam flow path as the primary supply for the TDMFWPs is not increased because the design and important to safety functions of the LP and HP components (governor and stop valves) remain the same and there are no significant impacts to the applicable accident analysis.

   Since the loss of normal feedwater is assumed and has no role in mitigation, the proposed activity does not increase the consequences of the Loss of Normal Feedwater Flow Accident The identified malfunctions result in the failure of equipment (TDMFWP) that is assumed to fail as a precursor to the related accident. The consequences of identified malfunctions, therefore, do not increase.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the proposed activity involves components associated only with the turbine MFWP steam supply flow path. Component design is not affected other than setpoint changes and resulting primary vs. backup roles of redundant flow paths With

the proposed activity implemented, the type of functions (and malfunctions) are the same as in the existing design.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the proposed activity has no effect on equipment required by Technical Specifications.

DESIGN CHANGE

D20-1-98-301-001

DESCRIPTION

The purpose of this Design Change was to replace the current refrigeration type air dryer installed on the 1A Emergency Diesel Generator (EDG) starting air system, with two membrane type air dryers. Some other supporting equipment (e.g. filters, solenoid valves, etc.) were added. The setpoints of two pressure switches and a relief valve were changed to enhance the operation of the starting air system and the EDG itself. Rerouting and re-classification of piping were employed in accordance with the approved codes and standards

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the modification did not change any initiating conditions or events associated with any analyzed accident/transient, nor changed/affected the functions of the EDG to mitigate any accident/transient.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the modification did not have an adverse impact on the reliability of the EDG or any of the interfacing systems. Also, the modification did not introduce any new operational limitations for the EDG or its subsystems

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because Technical Specifications 3.8.1 and 3.8.2 ensure that a reliable source of emergency power is available This modification did not challenge the reliability or the availability of the EDG.

NUCLEAR WORK REQUEST

990231989-02

DESCRIPTION

The purpose of this Nuclear Work Request #990231989-02 is to complete necessary requirements for the removal of a floor plug at location elevation 401 N-10 for cleaning of tank 0WX01DC. The floor plug is considered part of the auxiliary building ventilation boundary and, therefore, removal of the plug needs to be evaluated.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety as previously evaluated in the UFSAR, is not increased because removal of the floor plug does not have any impact on the events which initiate a Loss Of Coolant Accident (LOCA). Removal of the floor plug with the administrative controls in place ensures any design base accident remains bounded by the existing off-site dose boundary analysis calculation.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the possibility of an accident or malfunction of a different type than previously evaluated in the IFSAR is not created because the affected structure provides a ventilation boundary for Auxiliary Building HVAC (VA) systems. Part of the design requirements for the VA system is to limit environmental conditions in various zones in conformance with requirements. The system controls radioactivity in the areas served by staging the supply air from the clean areas to the areas of greater potential contamination. Plant operation is not changed No new failure modes are introduced.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the margin of safety, as defined in the basis of the Technical Specifications, is not reduced because the change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety.

Tracking Number: BRW-SESV-2001-193

PROCEDURE REVISION

2BwOSR 3.6.2.1-2
2BwOSR 3.6.2.1-3

DESCRIPTION

The purpose of this Procedure Revision was to revise the peak containment pressure corresponding to the revised calculated peak containment pressure at the uprated power level. Also revised the containment allowable leakage corresponding to the revised Pa at the uprated power level.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased since the procedure performs a test of the containment air locks (equipment and emergency). There is no effect on the operation or control of any UFSAR described SSCs.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not because the procedure performs a test of the containment hatches. The revision to the procedure involves changing the calculated peak containment pressure and the allowable leakrate. The values have been calculated as part of power uprate.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the values used for peak containment pressure and allowable leakage have been evaluated to be acceptable as part of the power uprate project.

PROCEDURE REVISION

RP-BR-750

DESCRIPTION

The purpose of these Procedures Revisions was to incorporate a calibration process for the AMS-4 monitor. This process merely replaces the service that is currently provided by an off-site contractor. The calibration process does not affect how the instrument is used on site. A previous screening is still valid for assessing the impact of the presence and use of the instrument upon site systems, structures, or components.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the respiratory protection procedures are designed to provide training, fit testing, evaluation, selection and use of respiratory protection devices, as well as maintenance and care of respiratory protection devices. There are no plant systems, structures, or components affected by the radiological respiratory program.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the implementation of a different calibration process merely replaces an off-site service and does not change the way in which the instrument is used at the site.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because this activity does not affect any parameters upon which the Technical Specifications are based.

UFSAR REVISION

UFSAR Draft Revision Package 8-111

DESCRIPTION

The purpose of this UFSAR Revision was to revise Section 5.4.1.3.10 of the UFSAR in order to ensure that the statement regarding the high flow alarm setpoint for RCP seal leakoff is consistent with vendor's recommendations and the normal expected range of operation

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because this activity revises the UFSAR information regarding the RCP #1 seal leakoff high flow alarms setpoint and does not change any initiating conditions nor does it change the normal operation of the RCP seal system. The revision, as stated in step 3 above, allows operation of the RCP seal with seal leakoff flow >6 gpm and <8 gpm. Present alarm setpoint is 4.8 gpm for all RCPs except 2A, which has a setpoint of 5.5 gpm, a value that is below the Westinghouse guidelines for RCP operation with a damaged RCP #1 seal The guidelines require initiation of orderly plant shutdown with seal leakoff flow >6 gpm and <8 gpm. After the implementation of this change the operator will still be capable of determining #1 seal leakoff flow for the affected RCP. Therefore, the operator will be capable of determining if the # 1 seal on the RCP is degrading so that appropriate actions can be taken to place the plant in a safe condition. Hence the probability of occurrence of a small break LOCA due to the failure of a RCP # 1 seal is not increased because the RCP will be operated within the guidelines set forth by Westinghouse Technical Bulletin ESBU-TB-93-01-R1 and operating procedures 1/2BwOA RCP-1.

   The consequences of a small break LOCA due to a RCP seal failure are not increased since the design or operation of the RCP is not negatively affected by this revision to the UFSAR The revision does not affect the operator's ability to determine and mitigate the effects of a RCP # 1 seal failure. Also, all safety related ECCS equipment is available for the purpose of mitigating the consequences of a small break LOCA in accordance with established operating procedures. Particularly, RCP seal integrity will be maintained since the RCP will still be operated within established guidelines (Westinghouse Technical Bulletin ESBU-TB-93-01-R1)

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the revision to the UFSAR and operation of 2A RCP with a #1 seal leakoff flow high alarm setpoint of 5.5. gpm will not create the possibility of an accident or transient of a different type than any previous evaluated since this revision only allows the operation of RCPs with a high seal flow that is within the range recommended by the RCP vendor. The present 2A RCP # 1 seal leakoff flow value of 5.5 gpm is less than 6.0 gpm. During plant operation the opèrator is required to shutdown the plant if they # 1 seal leakoff flow is outside the operating range of (0.8 – 6.0 gpm) High leakoff flow is indicative of a possible RCP # 1 seal failure, possibly resulting in a small break LOCA that was already considered in the safety analysis. Therefore failure of the RCP or occurrence of a transient,

as a result of operating with seal leakoff flows consistent with those recommended by the vendor, is highly unlikely. The proposed change merely provides basis for the accuracy of the display instrumentation in tables 7.5-1 and 7.5-2. The instrumentation only provides display function and does not impact any equipment or function Therefore, the probability of an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because Technical Specification 3.5.5 establishes a restriction on Reactor Coolant Pump (RCP) seal injection flow and limits the amount of ECCS flow that would be diverted from the injection path following an accident. With the revision to the high flow alarm for seal leakoff flow, the seal injection flows to the RCPs will still be limited to the specified ranges established in Fig 3.5.5-1 of the Technical Specification. Therefore, the margin of safety as described in the Technical Specification basis B.3.5.5 is not reduced.

   Technical Specification 3.4.13 specifies the amount of RCS identified leakage allowed during plant operation. This leakage does not include RCP controlled leakage. The margins specified here are not reduced because this revision to allow greater RCP seal leakoff flow does not affect this margin.

DESIGN CHANGE PACKAGE

9900582
9900583

DESCRIPTION

The purpose of these Design Change Packages was to replace Units 1 and 2 Reactor Cavity Nozzle Inspection Cavity covers The existing covers are heavy and hard to decontaminate. The replacement covers will be light enough to remove and replace without using the Containment Polar Crane. They will also be easier to decontaminate. This modification will save time and associated radiation dose for the removal and replacement of the covers.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the new covers are passive in nature and have no direct interface with any plant equipment. The decrease in mass associated with the new covers has been evaluated for effects on the containment heat sink and found to be acceptable The reduction in radiation shielding provided by the new covers has been evaluated and found to be acceptable. The new covers are qualified for the same loads as the original covers Installation and removal of the covers is procedurally controlled. No new failure modes are introduced, and the consequences of a new cover failure are the same as for the original covers.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the new covers are passive in nature and have no direct interface with any plant equipment. The reduction in radiation shielding provided by the new covers has been evaluated and found to be acceptable. The new covers are qualified for the same loads as the original covers. Installation and removal of the covers is procedurally controlled. No new failure modes are introduced.

   The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because Technical Specification 3.9.7, Refueling Cavity Water Level is the only applicable Technical Specification. The margin of safety is no reduced by this change because the new covers are qualified for the same loads as the original covers Installation and removal of the covers is procedurally controlled, and no new failure modes are introduced.

TECHNICAL SPECIFICATION BASES REVISION

(B 3.5.1)

DESCRIPTION

The purpose of this Technical Specification Bases Revision was to revise the Bases Section B 3.5.1, "Accumulators" to clarify the assumptions of the current LBLOCA and SBLOCA Analysis of Record (AOR).

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the Technical Specification Bases revisions proposed do not create any condition that would challenge the design or operational limitations of the affected equipment on the ECCS system. The proposed activity will not increase the probability of occurrence of a malfunction of equipment important to safety. The ECCS system remains capable of performing it's design accident mitigation function No functional changes operational modes, or failure mechanisms are introduced by the proposed revision, therefore, the consequence of a malfunction of equipment important to safety will not be increased.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the proposed Technical Specification Bases revision does not change the function of the ECCS system and affected components. There are no direct or indirect impacts on interfacing systems and components Since no new failure modes are created by the changes, there will be no malfunction of equipment of a type different from those evaluated in the UFSAR

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the ECCS system calculated cooling performance must meet the acceptance criteria as specified in 10CFR50.46(b). Approved modeling methods must include supporting justification to show that the analytical technique utilized realistically demonstrate the behavior of the RCS and ECCS systems during a postulated accident. Comparisons are made to applicable experimental data and uncertainties in the analysis method and inputs Once identified, results are compared to the ECCS acceptance criteria with a high level of probability that the limits will not be exceeded If it is determined that inputs (assumptions) were not incorporated, or may have resulted in error, corrections are performed and the new results are compared to the 10CFR50.46(b) Acceptance Criteria. Errors are not considered to be significant if the resulting Peak Cladding Temperature (PCT) does not change collectively or individually by more than 50°F. These errors are reported to the NRC annually.
   The PCT penalty for the water injection delay as described by this proposed activity has resulted in a 20°F penalty for Unit 2 only. This penalty combined with previously identified input parameter errors have increased the total PCT error to 46°F, and thus a new Unit 2 SBLOCA PCT value of 1871°F. The resultant PCT change remains "insignificant" and well below the acceptance criteria limit of 2200°F as described in 10CFR50.46 (b). Therefore, the Margin of Safety in NOT reduced.

SETPOINT CHANGE

DESCRIPTION

The purpose of this Setpoint Change was to lower the alert alarm setpoint on the 1RE-PR008 and 2RE-PR008 per BwRP 5820-5 due to a background decrease since the last setpoint calculation Revise BwVP RM80-3-1PR08 , Revision 4, and BwVP RM80-3-2PR08, Revision 3, to reflect the new Alert Alarm setpoints

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the alert alarm setpoint provides for an early indication of an upward trend on the radiation monitor which can aid in confirming a steam generator tube leak or rupture. The alert alarm setpoint has no affect on plant equipment or operation. The function of the radiation monitor remains unaffected by the alert alarm setpoint change. Changing the alert alarm setpoint will not change the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the setpoint change affects the monitor's software, it is not a physical adjustment to the monitor. The alert alarm setpoint provides for an early indication of an upward trend on the radiation monitor which can aid in confirming a steam generator tube leak or rupture. The alert alarm setpoint has no affect on plant equipment or operation. The function of the radiation monitor remains unaffected by the alert alarm setpoint change.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because steam generator blowdown radiation monitor alert alarm setpoints are not described in the basis of any Technical Specificaiton.

Safety Evaluation Summary Form

Tracking No. BRW-SESV-2001-217
Activity No. Offsite Dose Calculation Manual (ODCM), Chapter 10, Revision 3

DESCRIPTION:

The ODCM is being revised to remove the references to the interlock function associated with the 2RE-PR027 radiation monitor. This change is required due to implementation of Design Change D20-2-96-308 which removes the hi-radiation interlock signal from radiation detector 2RE-PR027 which starts the Off-Gas Exhaust Fan (0OG01C) due to abandonment of the off-gas filter unit (OGFU).

SAFETY EVALUATION SUMMARY:

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

The occurrence of steam generator tube leakage is governed by factors such as method of fabrication, metallurgy, chemistry, etc. The operation or non-operation of OFGU has no effect on the probability of occurrence of tube leakage. Calculation BRW-99-0468-M determined that the increase in thyroid dose associated with the abandonment of the off-gas filter unit is insignificant and within federal limits for primary to secondary leakage assumed to determine compliance with 10CFR50 Appendix I.

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

The OFGU is designed for normal operation with steam generator tube leakage. The system is normally in standby with exhaust gases bypassing the OFGU. The proposed change, therefore, will not impact the operation of the steam jet air ejectors, gland exhausters, or hogging pump function to support maintaining a vacuum in the main condenser. In addition, the proposed change does not affect the safety function of any systems or components. Reliance on the OGFU to maintain off-site dose within established limits is not necessary. Actions will be initiated prior to challenging these limits based on Technical Specification requirements and the permanent abandonment of the OGFU will not result in increased risk to the general public.

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

Since plant operation will continue to be in accordance with the controls of the ODCM and doses will be maintained within the criteria of 10CFR50 Appendix I and 10CFR20, there is no reduction in the margin of safety.

UFSAR REVISION

UFSAR Draft Revision Package 8-130

DESCRIPTION

The purpose of this UFSAR Revision (DRP) was to revise section 12.3.4.1 to clarify the frequency of calibration for safety-related, LCOAR, and TRM area radiation monitors as well as the remaining radiation monitors

SAFETY EVALUATION SUMMARY:

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the proposed UFSAR change does not change any initiating conditions or events associated with any accidents or transients It does not change the normal operation of the plant and system function

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the proposed change merely clarifies the frequency of calibration based on the safety significance and historical performance of the radiation monitors.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because there is no impact on the Technical Specifications, especially those listed in Step 5. The proposed change merely clarifies the frequency of calibration based on the safety significance and historical performance of the radiation monitors. Therefore, the margin of safety is not reduced.

Safety Evaluation Summary Form

Tracking No. BRW-E-2001-225
Activity No. _____

DESCRIPTION:

Braidwood Unit 1 Cycle 9 Core Operating Limits report sequence 7.

SAFETY EVALUATION SUMMARY:

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

   The COLR revision does not affect the existing safety analysis limits. The methodologies do not affect the normal plant operating parameters, the safeguards systems actuations, the accident mitigation capabilities of any SSCs, nor does it create conditions more limiting than those assumed in safety analysis.

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

   The demonstrated adherence to the standards and criteria in the COLR preclude new risks to components and systems that could introduce a new type of accident.

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

   All parameters in the COLR are verified in the core design process to maintain the margin of safety. The changes to the COLR reflect these verified values.

Safety Evaluation Summary Form

Tracking No. BRW-E-2001-226
Activity No. _____

DESCRIPTION:

Braidwood Unit 2 Cycle 9 Core Operating Limits report sequence 5.

SAFETY EVALUATION SUMMARY:

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

   The COLR revision does not affect the existing safety analysis limits. The methodologies do not affect the normal plant operating parameters, the safeguards systems actuations, the accident mitigation capabilities of any SSCs, nor does it create conditions more limiting than those assumed in safety analysis.

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

   The demonstrated adherence to the standards and criteria in the COLR preclude new risks to components and systems that could introduce a new type of accident.

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

   All parameters in the COLR are verified in the core design process to maintain the margin of safety. The changes to the COLR reflect these verified values.

UFSAR REVISION

DRP 8-158

DESCRIPTION

Under the DBI program for Byron and Braidwood stations, several discrepancies were found in the UFSAR. Some of these discrepancies will be corrected via DRP 8-158 in the next revision of the UFSAR as follows:

Per the DBI Open Item Number 1293, several parameters such as Ultimate Stress, Yield stress, Strength Coefficient (K), Hardness Exponent (n) and number of specimens tested for A-I 06 Grade B material will be corrected. Per Open Item Number 64, the method of combining the seismic response will be corrected. Per Open Item Number 2577, the load combinations will be revised to match Westinghouse Specifications 955926 and 966927. Per Open Item Number 2836, the dimensions for Category I Buildings will be revised to match their respective design drawing dimensions.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR is not increased because this change, (administrative and editorial in nature) does not impact any design features or adversely affect the ability to achieve and maintain safe shutdown of the plant. The UFSAR change will reflect the present designed bases for Byron/Braidwood Stations.

2. The possibility of an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because the change does not impact any equipment important to safety or affect safe shutdown capability. The changes are administrative and editorial in nature and reflect the design basis of the plant.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because all Technical Specification requirements were met. The changes are administrative and editorial in nature and reflect the design basis of the plant.

50.59 REVIEW COVERSHEET FORM

·tion: Braidwood

50.59 Evaluation Number: BRW-E-2001-230 **Revision Number:** 0

Activity/Document Number: Steam Table Software Package - SE0059 **Revision Number:** 0

Title: Steam Table Software Package

Description of Activity:

A Steam Table Software Package (STSP) is described/evaluated here and is intended for NGG wide use, on the PC Computer of any Engineer needing thermodynamic properties of steam or water during their normal work activities. The STSP is intended to be used for any Engineering function, including safety related functions.

The industry standard for thermodynamic properties for steam or water is the ASME Steam Tables. In this publication, water and steam property tables are provided. In addition, this publication provides a mathematical method which is intended to be used to calculate the values shown in the tables. The mathematical methods are shown in the ASME Steam Tables as Appendix 1, "The 1967 IFC Formulation For Industrial Use".

The STSP utilizes these ASME mathematical methods to provide thermodynamic properties for sub-critical steam or water. The sub-critical region covers any water or steam conditions encountered at a nuclear power plant.

Reason for Activity:

Several Engineering software applications are under consideration for development.

As an example:

A Hand Calorimetric Application.

This application is intended to be used in the Control Room to provide the Operator with calorimetric values in the event of a Plant Process Computer unavailability. It is intended to replace the current manual, laborious and error prone calculation method. In addition, this tool will be useful in performing independent verifications of calorimetric values provided by the main Calorimetric Software installed on the Plant Process Computer and to perform "what-if" evaluations on calorimetric related issues.

In order to develop this and other applications, a suitable and approved steam tables software package must be available for use in these applications. The STSP will enable these applications to obtain the necessary steam and water properties through the use of subroutine calls to the STSP.

Additionally, many Engineering assignments require the use of steam or water properties. These assignments are often safety related. As an aid to the Engineers performing these tasks, to avoid tedious interpolations, a suitable steam tables software package should be available to meet these requirements. The STSP includes a stand alone desktop application for the quick determination of steam or water properties.

Station: **Braidwood**

59 Evaluation Number: **BRW-E-2001-230** **Revision Number: 0**

Activity/Document Number: **Steam Table Software Package - SE0059** **Revision Number: 0**

Title: **Steam Table Software Package**

Effect of Activity:

The steam and water thermodynamic properties obtained from the STSP have been verified and validated (V&V) to be identical to those obtained from the industry standard ASME Steam Tables. The V&V process was performed as described in a Verification and Validation Plan contained in the STSP Software Management Plan, ID number SE0059. The Software Management Plan is required and controlled by NSP-CC-3021, Control of Computer Software and Services.

Since the steam and water thermodynamic properties obtained from the STSP have been demonstrated to be identical to those obtained from the ASME Steam Tables, the software is considered equivalent to the ASME Steam Tables and suitable for use in any safety related function associated with Exelon Nuclear Generating Stations. In addition, no adverse effects will result from the use of the STSP as an alternate to ASME Steam Table look-ups.

Summary of Conclusion for the Activities 50.59 Review:

Since the steam and water thermodynamic properties obtained from the STSP are demonstrated to be identical to those obtained from the ASME Steam Tables, no adverse effects will result from the use of the STSP as an alternate to ASME Steam Table look-ups.

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-2001-230**	**Rev.**	**0**
	50.59 Validation	**50.59 Validation No.**		**Rev.**	

DESIGN CHANGE PACKAGE

9900676

DESCRIPTION

The purpose of this Design Change Package (DCP) was to revise the setpoints of pressure switch 2PSDG094A. This device senses the pressure in Starting Air Receiver 2DG01SA-TD and controls the cycling (start/stop) sequence for the 2A Diesel Generator (DG) Number 2 Air Compressor (2DG01SA-B) and 2A DG Number 2 Air Dryer. Currently at a pressure 240 psig decreasing, pressure contacts start the associated air dryer and initiate a 90 second time delay. After the 90 second time delay, the compressor starts When receiver pressure reaches 250 psig both the compressor and dryer stop This change will lower the cycle band such that the compressor cycle start/stop cycle starts at 225 psig decreasing and stops at 235 psig. Other than the cycle setpoint, the DCP will not impact the starting logic for the 2A DG Number 2 Air Dryer and Air Compressor for an emergency mode or test mode start of the 2A DG. Document Review Package DRP 9-052 changes references to the cycling setpoint as listed in UFSAR Section 9.5.6.1.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the proposed activities do not change any initiating conditions or events associated with any accident or transient, nor do they change the normal operation of the Diesel Generators The changes to implemented under this SSCR do not adversely affect Diesel Generator reliability or availability. The Diesel Generators remain capable of performing their intended safety function as required to mitigate the consequences of the affected accidents

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the changes implemented do not have adverse impact on the reliability of the 2A DG, nor do they impact the reliability of any interfacing system The changes do not introduce any new operational limitations for the affected engine subsystems, nor do they challenge the availability of the 2A DG. The changes do not impact 2A DG performance in any way. There is no interaction between these changes and any other equipment important to safety.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the implementation of the changes/activities do not challenge the reliability or availability of the Diesel Generators as a source of AC power, and therefore does not reduce the margin of safety as described in the Bases of Technical Specifications and supporting SAR documents

UFSAR REVISION

UFSAR Draft Package 9-030

DESCRIPTION

The purpose of this UFSAR Revision was to revise Section 6.3.2.2 and UFSAR Table 6.3-2 to reflect the installation of modification #D20-2-98-341. This modification replaced the Crosby JRAK model relief valves on the discharge of the Safety Injection (SI) pumps with OMNI 900 models. The specific valves are 2SI8853A (SI pump 2A discharge), 2SI8853B (SI pump 2B discharge), and 2SI8851 (Common discharge header from the SI pumps to the RCS loops cold legs). The setpoint of the relief valves was changed from 1750 psig to 1810 psig Evaluation BRW-V-2001-241 is a validation of Safety Evaluation BRW-SE-1999-395

Note – Preparation of this validation document was started before the March 13, 2001 implementation date for the new 50.59 Rule. Thus, completion of this validation was done per the old 50.59 procedures

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the relief valves replacement does not change or impact the initiating conditions or events which result in design basis accidents. The integrity of the Reactor Coolant or Main Steam pressure boundary is not degraded. The increase in relief capacity of the new relief valves does not impact the pressure integrity of the Recycle Hold-up Tanks (RHUTs) and will not increase the probability of failure of these tanks.

   The function or operation of the Emergency Core Cooling System (ECCS) is not affected. The higher set pressure will cause a higher discharge flow rate due to the higher differential pressure across the valve However, the discharge of these valves is routed to the RHUTs and there is no increase in the potential of uncontrolled release of contaminated fluid. The integrity of the Safety Injection (SI) system pressure boundary and injection flowpaths are not degraded. The design pressure of piping exposed to the higher pressure has been increased consistent with licensing commitments to ASME criteria

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the relief valves replacements do not change the function of the SI System and affected components. There are no direct or indirect impacts on interfacing systems and components

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the proposed changes maintained the integrity ad the function of the SI System flowpaths relied upon to perform injection and long term cooling functions for the ECCS. The acceptability of the increase in system design pressure has been verified by calculaiton.

SPECIAL PROCESS PROCEDURE

SPP 01-01, Rev. 0
SPP 01-02, Rev. 0

DESCRIPTION

The purpose of this Special Process Procedure was to direct the online implementation of Braidwood Unit 1 and Unit 2 power ascension from a core power level of 3411 MWT to the "Mini" uprated power level which represents an increase of 1% steam flow from the former 100% rated thermal power level. The uprated power ascension is accomplished in controlled steps as follows:

- While holding the unit at the pre-uprate full power, with Advanced Measurement and Analysis Group (AMAG) feed flow scaling factors in place, the Reactor Rated Thermal Power computer Point (K8144) will be changed from 3411 MWt to 3586 6 MWt resulting in an instant Calorimetric indicated change from 100% to 95.1% · Various changes to nuclear, RCS, BOP, and Radiation Monitoring instrumentation will be implemented at this power level.
- Power will be raised to 96.1% indicated power level which represents an increase of 1% steam flow from the former 100% rated thermal power level.
- After a one-hour soak period at this new power level, plant systems and equipment will be monitored to confirm continued operation at the "Mini" uprated power level

Following completion of this procedure, power will be limited to the "Mini" uprated power level until the planned HP turbine replacement is complete during A1R09

The Mini Uprate Power Ascension procedures do not direct configuration activities program impact, procedure impact, process computer impact and training impact reviews and updates These activities are directed by the Power Uprate Master DCP 9900597 "Braidwood Unit 1 Power Uprate Implementation" and DCP 9900598 "Braidwood Unit 2 Power Uprate Implementation".

SAFETY EVALUATION SUMMARY:

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the Mini Power Uprate Ascension procedures identify the expected operation parameters for power ascension evolution. In addition, acceptance limits are specified for ensuring the plant parameters are maintained within the power uprate analyses limits with margin Level 1 parameters will be monitored continuously and power ascension will be monitored after power ascension evolution and reviewed against the acceptance The Power Uprate Ascension procedures do not require SSCs to perform functions outside their design basis and therefore do not introduce any new potential failures or operating transients. All operations during power ascension are performed in accordance approved station procedures and Technical Specification requirements as applicable to the Power Uprate. During implementation of setpoint and scaling changes, the procedures provide the necessary steps and Notes to ensure the plant configuration is maintained while at power and the necessary

operating indications or manual operations are available and identified to opeations A Heightened Level of Awareness (HLA) briefing will be performed with all personnel involved with the power ascension testing. All plant and equipment operation and surveillance will be performed in accordance with approved plant procedures

The power ascension will not result in any system or component required for accident or transient mitigation to be operated, shutdown, or isolated such as to impact its ability to perform its safety-related mitigating functions. The power ascension from the current license power level to the Mini-uprate power level does not impair any safety-related SSC's performance during accident or transient conditions. Therefore, the proposed activity will not increase the probability of occurrence or the consequences of any accident or a malfunction of equipment important to safety during power ascension.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the configuration, operation and accident response of the Braidwood Unit 1 and 2 systems and components are unchanged by operation during the power ascension. Analyses of transient events as a result of the uprate have confirmed that no transient event results in new sequence of events that could lead to a new or different accident scenario The effect of operation at the uprated power conditions on plant equipment was evaluated. No operating mode, equipment line-up, or equipment failure was identified that could result in a new or different accident or malfunction The proposed activity does not result in any system or component to operate outside the power uprate analyses assumptions or initial conditions, or operating mode, equipment line-up, or equipment failure that could result in a new or different accident or transient than that analyzed for the power uprate.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because as part of the Power Uprate Project, the entire licensing basis of the plant was reviewed for impacts due to the increase in licensed core thermal power. Items reviewed included the entire UFSAR, pending UFSAR changes, SER, Technical Specifications, Fire Protection Report (FPR) and licensing correspondence. These UFSAR and FPR changes will be processed under DRP 9-009, DRP 9-010 and DRP 9-020 and FPR 20-004. DRP 9-009 processes the UFSAR changes to all the chapter 15 accident analyses DRP 9-010 processes the UFSAR changes to the containment system on chapter 6, while DRP 9-020 processes the changes for the remaining UFSAR chapters Review of the Technical Requirements Manual (TRM) and Technical Specification Basis also identified the basis B.3 6.8 "Hydrogen recombiner" is affected by Power Uprate These TS changes were identified in the Licensing Amendment Request (LAR). The Proposed activities are enveloped by the analysis as described in the Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations (RS-0038, dated 7/5/2000).

50.59 REVIEW COVERSHEET FORM

BRW-E-2001-256

.ation: BRAIDWOOD

Activity/Document Number: Unit 1 - DCP 9900677 Unit 2 - DCP9900679 **Revision Number:** 0

Title: Revenue Metering for System Auxiliary Transformers (SAT)

Description of Activity:

These DCPs will install outdoor free-standing combined current and voltage (potential) transformers (CT/PT) in the station switchyard for metering station electrical power usage. These combined CT/PT transformers (3 per Unit) will be installed in-line on the345 KV feed to the System Auxiliary Transformers (SATs). This DCP implements changes in the switchyard, and impacts the offsite electrical feeds to the SATs which are the preferred source of power to the 4.16KV ESF buses.

Reason for Activity:

The CT/PTs are being installed in the 345 KV feed lines to the SATs in order to meter power consumption of the SATs and their loads. The ComEd Interconnection Agreement, FERC open access transmission rule and ICC regulations require metering at the point of interconnection (high side of transformer) to the ComEd transmission system.

Effect of Activity:

The effect of these changes will be to physically install and electrically connect combined CT/PTs in the 345 KV lines feeding the SATs. There will be no effect on plant equipment or plant operation when the CT/PTs function as designed. The failure of a CT/PT could result in the loss of offsite power (LOOP) to the SATs and the 4.16 KV ESF buses. However, the effect and consequences of this failure are no different than the result of any failure which causes the loss of this feed to the SATs and are bounded by existing UFSAR analyses. In addition, due to the complexity of the CT/PTs compared to the 345 KV line, the likelihood of a malfunction has increased. This evaluation has determined that there will not be more than a minimal increase in the likelihood of a malfunction.

These CT/PT combination units are SF6-gas-insulated and required to be monitored for gas pressure until the low gas pressure alarms are connected to alarm at the Bulk Power Operations center. Therefore, revisions to the weekly switchyard surveillance and operator training will be required as a result of these changes.

Braidwood Station Procedure 0BwOS SY-W1 "Unit Common 345 KV Switchyard Weekly Surveillance" will be revised to include surveillance of the CT/PTs. Operators will be trained to visually check the SF6 gas pressure reading on the gauges of the CT/PT units to verify gas pressure is within an acceptable range. Training packages are assigned for this purpose. A 50.59 evaluation has been performed since the failure of a CT/PT unit could result in a LOOP. Effects and consequences of a LOOP are bounded by current UFSAR accident analyses, which determined that a LOOP will not prevent the safe shutdown of the reactor or mitigation of accidents or accident consequences.

Summary of Conclusion for the Activities 50.59 Review:

The proposed activity is a design change to the plant which affects SSCs that perform a design function. Since the proposed activity was considered adverse due to the potential reduction in reliability resulting from the installation of the CT/PT combination units, a *50.59* evaluation was performed.

Failure of a CT/PT unit can result in the Loss of Offsite power to the SATs, which ultimately may result in a loss of offsite power to the 4.16kv ESF buses. However, it does not affect the onsite Emergency DGs capability to power 4.16kv ESE buses for safe shutdown and post accident recovery. This event is already analyzed in the UFSAR accident analyses which determined that it will not prevent safe shutdown of the reactor or adversely affect mitigation of an accident or accident consequences. Although a new component susceptible to failure is being installed, the result of a malfunction or failure is not different than previously analyzed.

The conclusion of the evaluation was that the activity does not require NRC approval and the proposed changes may be implemented per applicable procedures.

BRW-E-2001-256

Station: BRAIDWOOD

Activity/Document Number: Unit 1 - DCP 9900677 Unit 2 - DCP9900679 **Revision Number:** 0

Title: Revenue Metering for System Auxiliary Transformers (SAT)

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-2001-256**	**Rev.**	**0**
	50.59 Validation	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

Station: Braidwood

Activity/Document Number: Implement Procedure Revision / CC-AA-411 **Revision Number:** 1

Title: Maintenance Specification: Requirements for Use of Alternative Insulation Outside the Containment Building

Description of Activity:

Revision 1 to Maintenance Specification CC-AA-411 adds clarification that the insulation used on stainless steel pipe shall have material on the "hot-side" of the insulation that complies with RG 1.36 requirements. This revision specifically states that the "hot-side" of the Heatainer covers is SIL 1700 instead of TEF 1700.

See sections 4.1.2, 5.1.5, and 6.1.1.

Also, a clarification is added in Section 5.1.6 with respect to Mark 1 BWR Containment Piping, which does not apply to Braidwood and need not be discussed further.

Reason for Activity:

To prevent leachable chlorides & fluorides from the replacement insulating materials from coming into contact with stainless steel piping in compliance with Reg. Guide 1.36 requirements.

Effect of Activity:

This clarification will prevent the use of insulating material "hot-side" covers containing leachable chlorides & fluorides from being used on stainless steel piping at Braidwood Station. This will help prevent chloride-fluoride induced stress corrosion in austenitic stainless steel piping.

Summary of Conclusion for the Activities 50.59 Review:

Adding the clarification with respect to using materials in compliance with Reg. Guide 1.36 has no effect on plant operating systems, Technical Specifications, or USFAR documents.

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

Forms Attached: (Check all that apply.)

[ ]	**Applicability Review**				
[ ]	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
[ ]	**50.59 Evaluation**	**50.59 Evaluation No.**		**Rev.**	
X	**50.59 Validation**	**50.59 Validation No.**	BRW-V-2001-260	**Rev.**	0

50.59 REVIEW COVERSHEET FORM

LS-AA-104-1001
01/11/01

ion: Braidwood

Activity/Document Number: Implement Procedure Revision / CC-AA-411 **Revision Number:** 1

Title: Maintenance Specification: Requirements for Use of Alternative Insulation Outside the Containment Building

Description of Activity:

Revision 1 to Maintenance Specification CC-AA-411 adds clarification that the insulation used on stainless steel pipe shall have material on the "hot-side" of the insulation that complies with RG 1.36 requirements. This revision specifically states that the "hot-side" of the Heatainer covers is SIL 1700 instead of TEF 1700.

See sections 4.1.2, 5.1.5, and 6.1.1.

Also, a clarification is added in Section 5.1.6 with respect to Mark 1 BWR Containment Piping, which does not apply to Braidwood and need not be discussed further.

Reason for Activity:

To prevent leachable chlorides & fluorides from the replacement insulating materials from coming into contact with stainless steel piping in compliance with Reg. Guide 1.36 requirements.

Effect of Activity:

This clarification will prevent the use of insulating material "hot-side" covers containing leachable chlorides & fluorides from being used on stainless steel piping at Braidwood Station. This will help prevent chloride-fluoride induced stress corrosion in austenitic stainless steel piping.

Summary of Conclusion for the Activities 50.59 Review:

Adding the clarification with respect to using materials in compliance with Reg. Guide 1.36 has no effect on plant operating systems, Technical Specifications, or USFAR documents.

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

Forms Attached: (Check all that apply.)

[ ]	**Applicability Review**				
[ ]	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
[ ]	**50.59 Evaluation**	**50.59 Evaluation No.**		**Rev.**	
X	**50.59 Validation**	**50.59 Validation No.**	BRW-V-2001-261	**Rev.**	0

50.59 REVIEW COVERSHEET FORM

S‹ .ı: Braidwood

Activity/Document Number: DCR 990907 Revision Number: 0

Title: Revise drawings to reflect correct time delay relay setting of 15 seconds for 1AF01J-K11.

Description of Activity:

DCR 990907 revises the setting on time delay relay 2AF01J-K11 shown on drawings 20E-2-4030AF12, 20E-2-4468, and 20E-2-4469B from 10 seconds to 15 seconds.

Reason for Activity:

SSCR 87-036 was initiated per BwAP 1610-4T2 to increase the setting of time delay relay 1AF01J-K11 from 10 to 15 seconds. As a result of this SSCR both the unit 1 and unit 2 time delay relays were revised. DCR 990907 documents the change to the unit 2 time delay relay. The unit 1 change was previously documented on DCR 990263.

Effect of Activity:

These drawing changes correct the time delay relays that are shown on the unit 2 drawings to be consistent with current relay settings and procedures.

Summary of Conclusion for the Activities 50.59 Review:

drawing changes listed above require a 50.59 screening per procedure. The attached screening shows these drawing changes as requiring a complete 50.59 evaluation. The drawing changes do not adversely effect UFSAR described design functions, UFSAR ıgn function performance or control, design bases evaluation methodology, or tests and experiments inconsistent with analysis or descriptions in the UFSAR. The drawing changes do not require a Tech Spec change.

Attachments:

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

Forms Attached: (Check all that apply.)

[ ]	**Applicability Review**				
[ ]	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
[ ]	**50.59 Evaluation**	**50.59 Evaluation No.**		**Rev.**	
X	**50.59 Validation**	**50.59 Validation No.**	**BRW-V-2001-262**	**Rev.**	**0**

TECHNICAL REQUIREMENTS MANUAL

TRM Revision 01-004

DESCRIPTION

The Braidwood Unit 1 Pressure-Temperature Limit Report (PTLR), an attachment to the Braidwood Technical Requirements Manual (TRM), and BwCB-1 Figures 27 and 29 are being revised.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the revised Unit 1 Pressure and Temperature Limits Report (PTLR) maintains the same limits on Reactor Pressure Vessel pressure, as a function of temperature, as previously established That is, the Unit 1 heatup and cooldown operating provisions are unchanged by this revision. Also, the low temperature overpressure protection (LTOP) system limits, including the pressurizer power operated relief valve (PORV) setpoints, are unchanged by this revision.

   The net result of the changes to the PTLR from material property updates, uprated fluence levels, and updated neutron cross section data is the modification of the time frame for which this report remains valid. There are no hardware or software changes which could modify the frequency of initiating events.

   Maintaining the Reactor Coolant System (RCS) in the acceptable, analyzed, operating pressure - temperature range does not add to the number of challenges to the system. Therefore the frequency of transients or accidents which impact the integrity of the reactor coolant pressure boundary during low temperature operation, heatup, cooldown, criticality, normal operation, and leak or hydrostatic testing remains unchanged.

   These curves have been verified to maintain compliance with 10CFR 50 60 "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation", Appendix G to Part 50 "Fracture Toughness Requirements", and 10 CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events " Maintaining compliance with these requirements will not have a detrimental effect on RCS related SSCs but rather will assure that the SSCs remain within design This assures that the likelihood of a malfunction of an SSC that could challenge the Unit 1 RCS structural integrity has not increased with this change to the PTLR.

   The pressure - temperature curves are established with sufficient margin to accommodate postulated transient conditions of the RCS. The operation of the RCS within the acceptable limits of the revised P-T curves does not influence the creation of event initiators nor does it affect the severity of the radiological consequences of any analyzed event.

2 The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the operation of the RCS within the acceptable limits of the revised P-T curves does not have a detrimental effect on the RCS or interconnected / related SSCs. The revised curves do not reduce any operating margin and there are no new requirements or challenges to operating the RCS per the requirements of the revised curves. These revised curves meet all ASME Code and Regulatory requirements Therefore, because the changes to the Unit 1 P-T curves do not degrade the design of the RCS system in any way and do not change how the system performs or how it is procedurally controlled and operated, the changes will not create the possibility for an unanalyzed event

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the PTLR establishes the approved operating, pressure - temperature range for the RCS - a fission product barrier. The change to the Unit 1 PTLR does not modify this range. The limits (established by NRC approved methodology, have not been altered The change in the EFPY limit on the Unit 1 PTLR does not modify the criterion for the design of the RCS but rather establishes a date by which the curves would need reassessment Therefore the design basis limits of the RCS are not being exceeded or altered

DESIGN CHANGE PACKAGE

DCP 9900685

DESCRIPTION

The purpose of this Design Change Package (DCP) was to temporarily abandon currently install pressure switch 1PS-FW167 (which has failed) for valve 1FW009B. A like for like temporary pressure switch will be installed downstream of the hydraulic system fill valve on the valve actuator. The electrical leads from the new pressure switch will be landed in their place inside Junction Box 1FW05JB.

SAFETY EVALUATION SUMMARY·

1. The probability of occurrence or the consequences of an accident or a malfunction or equipment important to safety previously evaluated in the safety analysis report is not increased because the temporary pressure switch provides the identical function of the temporarily abandoned 1PS-FW167 switch. The installation of the temporary pressure switch does not affect the safety function of valve 1FW009B, which is to close on a containment isolation signal. The new switch has been functionally and seismically evaluated and determined to be acceptable.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the new temporary pressure switch performs the same function as 1PS-FW167 and is identical in form-fit-function.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the temporary pressure switch does not have any impact on the containment isolation requirements (TS 3.6.3) of valve 1FW009B. This TMOD does not affect normal valve function or its safety function (to close on containment isolation signal)

TECHNICAL REQUIREMENTS MANUAL

TRM Revision 01-004

DESCRIPTION

The purpose of this Technical Requirements Manual (TRM) Revision was to the Braidwood Unit 2 Pressure-Temperature Limit Report (PTLR), an attachment to the Braidwood Technical Requirements Manual (TRM), and BwCB-2 Figures 27, 28, and 29 are being revised.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the revised Unit 2 Pressure and Temperature Limits Report (PTLR) shifts the limits on Reactor Pressure Vessel pressure, as a function of temperature, to a more conservative value. That is, for a given temperature the revised PTLR requires a lower operating RCS pressure. However these new limits are established with an approved methodology (WCAP-14040) which assures compliance, with margin, with all Regulatory requirements. Maintaining the Reactor Coolant System (RCS) in an acceptable, analyzed, operating pressure - temperature range does not add to the number of challenges to the system. Since the PTLR still maintains sufficient margin to accommodate postulated transient conditions, the frequency of the occurrence of transients or accidents which could impact the integrity of the reactor coolant pressure boundary during low temperature operation, heatup, cooldown, criticality, normal operation, and leak or hydrostatic testing remains unchanged.

   These curves have been verified to maintain compliance with 10CFR 50.60 "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation"; Appendix G to Part 50 "Fracture Toughness Requirements"; and 10 CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." Maintaining compliance with these requirements will not have a detrimental effect on RCS related SSCs but rather will assure that the SSCs remain within design This assures that the likelihood of a malfunction of an SSC that could challenge the Unit 2 RCS structural integrity has not increased with this change to the PTLR

   This revision to the Braidwood Unit 2 PTLR maintains the structural integrity of the reactor vessel and the reactor coolant pressure boundary and provides protection against the occurrence of analyzed events such as large and small break loss of coolant accidents (LOCAs) Maintaining the proper design characteristics of the RCS will not negatively affect the level of radiological consequences of any evaluated event. Therefore these PTLR changes which provide for the continuing maintenance of RCS pressure boundary integrity as designed and analyzed assures that there is no change in the radiological consequences resulting from assumed events

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

the acceptable limits of the revised P-T curves does not have a detrimental effect on the RCS or interconnected / related SSCs The revised curves have an insignificant impact on operating margin and there are no new challenges to operating the RCS per the requirements of the revised curves These revised curves meet all ASME Code and Regulatory requirements. Therefore, because the changes to the Unit 2 P-T curves do not degrade the design of the RCS system in any way and do not change how the system performs or how it is procedurally controlled and operated, the changes will not create the possibility for an unanalyzed event.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the PTLR establishes the approved operating, pressure - temperature range for the RCS - a fission product barrier - well within the design basis limits for the RCS. The change to the Unit 2 PTLR does not modify this design basis limit. The PTLR curves, established by NRC approved methodology, meet the requirements of 10 CFR 50 Appendix G and Appendix G to the 1996 ASME Section XI code. These requirements assure RCS design limits are not exceeded with the changes in the reactor vessel material properties and increases in neutron fluence.

50.59 REVIEW COVERSHEET FORM

BRW-E-2001-301

Station: Braidwood

Activity/Document Number: BwVS 500-6 **Revision Number:** 6

Title: Low Power Physics Testing program with Dynamic Rod Worth Measurement

Description of Activity:

1. The primary revision in this activity is the changing of the control rod worth uncertainty from 10% to 7%. In order to facilitate this change, the following are revised:
   - Change the review criteria for rod worth testing to read "The sum of the measured worths are within 5.6% of the sum of the predicted worths"
   - Change the acceptance criteria to read 'The sum of the measured worths are ≥ 93% of the sum of the predicted worths"

   Safety Evaluation BRW-E-2001-301 Rev. 0 was generated to support this change. A Screening was also attached (does not have a number since it is for info only) to document the thought process which led to the generation of the Safety Evaluation.

2. As a result of Braidwood Tech Spec Amendment 111 (BDPS Elimination) and the implementation of the associated modification (D20-1-99-383), references to BDPS instrumentation/controls had to be revised such that the system could be operated as designed. A 50.59 Applicability Review form was attached to document the fact that a Screening was not required since this procedure change was required because of a Technical Specification revision.

3. Administrative limitations applicable to personnel in containment during Physics Testing were removed. These were included in the procedure to support the special testing requirements following steam generator replacement. Since these requirements are no longer applicable, they have been removed. Since this is considered a Managerial or administrative procedure change per the attached 50.59 Applicability Review form, a separate Screening was not required.

4. Various editorial changes were also included in this revision. The changes made as a result of lessons learned (from Byron and Braidwood) and changes in format will enhance the performance of this procedure.

Reason for Activity:

1. UFSAR Section 4.3.2.4 and Table 4.3-2 assume the use of 10% control rod worth uncertainty. As part of Power Uprate, the control rod worth uncertainty used in Shutdown Margin (SDM) calculations was changed to 7%. In order to comply with the assumptions made during the Power Uprate analysis, the surveillance that measures rod worths during Physics Testing is being revised.

2. As a result of Braidwood Tech Spec Amendment 111 (BDPS Elimination) and the implementation of the associated modification (D20-1-99-383), references to BDPS instrumentation/controls had to be revised such that the system could be operated as designed.

3. Administrative limitations applicable to personnel in containment during Physics Testing were removed. These were included in the procedure to support the special testing requirements following steam generator replacement. Since these requirements are no longer applicable, they have been removed.

4. Various editorial changes were also included in this revision. The changes made as a result of lessons learned (from Byron and Braidwood) and changes in format will enhance the performance of this procedure.

Effect of Activity:

1. During future Dynamic Rod worth testing, more restrictive requirements will need to be met for rod worth uncertainty. Recent history at Braidwood (and Byron) has shown that the new acceptance and review criteria can be successfully met.
2. BDPS controls will be operated as designed.
3. Administrative controls, which are no longer applicable, will be removed.
4. The editorial changes made as a result of lessons learned (from Byron and Braidwood) and changes in format will enhance the performance of this procedure.

Summary of Conclusion for the Activities 50.59 Review:

) proposed activity can be implemented without prior approval from the NRC since it is not considered a departure from a method of evaluation described in the UFSAR.

Note:
- 50.59 Safety Evaluation (BRW-E-2001-301) is used for Item 1 (change of the control rod worth uncertainty from 10% to 7%)
- Applicability review is used for Items 2 & 3 (BDPS elimination and Admin limitation removal)
- Item 4 is editorial change and is not applicable for 50.59 review

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.)

x	**Applicability Review**				
x	**50.59 Screening**	**50.59 Screening No.**	**N/A (screening attached for info only)**	**Rev.**	
x	**50.59 Evaluation**	**50.59 Evaluation No.**	**BRW-E-2001-301**	**Rev.**	**0**
	50.59 Validation	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

tion: Braidwood

Activity/Document Number: DCP 9900494 (SSCR 00-081) & DCP 9900511 (SSCR 00-098) Revision Number: 1

Title: Unit 1 and Unit 2 rescaling of f1(ΔI) penalty limits

Description of Activity:

The proposed activity will use the revised f1(ΔI) penalty limits for rescaling of the Delta Temperature / Temperature Average loops via DCP 9900494 (SSCR 00-081) for Unit 1 and DCP 9900511 (SSCR 00-098) for Unit 2 to be in accordance with the revised COLRs for Unit 1 and Unit 2 during implementation of mini-uprate to support operating cycles 9A.

Reason for Activity:

To be in accordance with revised Unit COLRs NFM0100052, Sequence number 0 (Unit 1) and NFM0100053, Sequence number 0 (Unit 2) to support power uprate changes during mid-May, 2001.

Effect of Activity:

The effect of rescaling the f1(ΔI) penalty limits will ensure that the clad stress criteria is not compromized.

Summary of Conclusion for the Activities 50.59 Review:

Rescaling of the f1(ΔI) penalty limits is supported by 50.59 screening validated. The COLR limits that were discussed in the validated 50.59 are being implemented by the DCP packages associated with this screening; therefore, the implementation of the COLR values are bounded by the original evaluation

Attachments:

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

Forms Attached: (Check all that apply.)

	Applicability Review		
	50.59 Screening	**50.59 Screening No.**	**Rev.**
	50.59 Evaluation	**50.59 Evaluation No.**	**Rev.**
X	**50.59 Validation**	**50.59 Validation No.** BRW-V-2001-0306	**Rev.** 0

50.59 REVIEW COVERSHEET FORM

BRW-E-2001-311

:ion: Braidwood/Byron

Activity/Document Number: UFSAR Change Package #DRP 9-058 **Revision Number:** 0

Title: Increase ESF Recirculation Loop Leakage Outside Containment

Description of Activity:

DRP 9-058 increases the permitted leakage for Engineered Safety Features (ESF) Equipment located outside containment. The specific increases to the leakage values given in UFSAR Table 15.6-13 are as follows:

- From 3,910 cc/hr to 15,249 cc/hr (about 4 gallons/hour) for Braidwood Station
- From 3,910 cc/hr to 13,294 cc/hr (about 3.5 gallons/hour) for Byron Station

The consequences from a Loss of Coolant Accident (LOCA) refer to radiation dose to plant personnel and offsite dose to the public. As stated in the Standard Review Plan, section 15.6.5 Appendix A, the radiological consequences from ESF equipment leakage outside containment are combined with the consequences of other fission product release paths (containment leakage) from a hypothetical LOCA. DRP 9-05 8 will also revise the UFSAR tables listed below to reflect the radiation exposure dose that corresponds to the increased ESF Equipment leakage outside containment:

- UFSAR Table 6.4-1, "Expected Dose to Control Room Personnel at Byron Station Following a Loss-of-Coolant Accident (LOCA)
- UFSAR Table 6.4-1, "Expected Dose to Control Room Personnel at Braidwood Station Following a Loss-of-Coolant Accident (LOCA)
- UFSAR Table 15.0-11, "Potential Offsite Doses due to Accidents", (Byron Station)
- UFSAR Table 15.0-12, "Potential Offsite Doses due to Accidents", (Braidwood Station)

Table 6.4-la, "Principal Assumptions Used in Control Room Habitability Calculations" for Byron and Table 6.4-la, "Principal Assumptions Used m Control Room Habitability Calculations" for Braidwood will also be revised to remove the 1 gal/hr assumption for ECCS leakage. This assumption will be replaced with "UFSAR Table 15.6-13". This change will (1) link tables 6.4-la with table 15.6-13 and (2) will eliminate the need to revise table 6.4-la each time a change is made to the ECCS leakage value is made. The "ECCS Leakage" wording in UFSAR Tables 6.4-la will be revised to "ESF Equipment Leakage". This is done to provide consistency with the terminology given in UFSAR Tables 6.4-1. The leakage limitations given in UFSAR Table 15.6-13 also apply to the Containment Spray System when it takes suction from the Emergency sumps following a Large Break LOCA (Refer to NUREG-0800, Standard Review Plan, Section 15.6.5, Appendix B).

Additionally, DRP 9-058 changes the cross-reference made to the table showing ESF equipment leakage from "Table 15.6-14" to "Table 15.6-13" in UFSAR section 6.4.4.1. The current cross-reference to Table 15.6-14 is incorrect. This change to section 6.4.4.1 is an editorial change; application of the 10 CFR 50.59 process to this change is not necessary.

Reason for Activity:

Provide a greater operating range without degrading the operation of any plant components.

The existing leakage values for ESF equipment outside containment are extremely restrictive. Exceeding the UFSAR values requires an evaluation for operability in relation to the resulting dose as compared to the appropriate limits (General Design Criterion 19 for the Main Control Room and 10 CFR 100 for the Exclusion Area Boundary and Low Population Zone). The ESF equipment leakage that would result in exceeding the regulatory limits is significantly higher than the maximum permitted leakage value of about 4 gallons/hour (For pre-uprate analyses nearly 1 gpm, Reference Operability Determination #99-029).

BRW-E-2001-311

Station: Braidwood/Byron

ivity/Document Number: UFSAR Change Package #DRP 9-058 **Revision Number:** 0

Title: Increase ESF Recirculation Loop Leakage Outside Containment

Effect of Activity:

Raising the allowed leakage for the ESF recirculation loop does not impact Plant operations. Leakage components will still be identified and corrective actions will be scheduled in accordance with the Corrective Action Program. The only difference will be that an operability assessment will be performed only if the actual leakage is above the new UFSAR leakage value.

Summary of Conclusion for the Activities 50.59 Review:

Raising the ESF Recirculation loop leakage outside containment to the values specified above will result in a Minimal Increase in dose consequences from a Loss of Coolant Accident (LOCA). The resulting Minimal Increase falls within the allowed increase from Document #NEI-96-07 Revision 1 as endorsed by the NRC via Regulatory Guide 1.187.

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.

ns 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-2001-311	**Rev.**	0
[ ]	**50.59 Validation**	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

BRW-V-2001-314

ation: Braidwood

Activity/Document Number: BRW-V-2001-314 Revision Number: 0

Title: Validation of BRW-E-2001-267

Description of Activity:

Revise BwOR WEST-260 "Top Nozzle Anchor Installation Field Procedure" to Rev. 1 to allow alternate guide tubes to be used for the anchor installation, and allow for the installation of additional anchors as necessary.

Reason for Activity:

Problems were encountered with installing Top Nozzle Anchors into the specified thimbles.

Effect of Activity:

The four locations specified in the procedure were selected as convenient symmetric locations. There was no technical reason to use these locations in place of any other four symmetric locations. Therefore, changing the locations does not change the ability of the anchors to support the fuel assembly load. Additional anchors would only increase the support.

Summary of Conclusion for the Activities 50.59 Review:

There are no issues with incorporating the new rev of the procedure.

tachments:

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

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☐	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
☐	**50.59 Evaluation**	**50.59 Evaluation No.**		**Rev.**	
X	**50.59 Validation**	**50.59 Validation No.**	BRW-V-2001-314	**Rev.**	0

Tracking Number: BRW-V-2001-337

PROCEDURE REVISION

SPP 01-002 Rev 1

DESCRIPTION

The SER BRW-E-2001-255 described the activities related to the mini power uprate procedures SPP 01-001, 01-002 revision 0 at Braidwood Ul and U2. These procedures direct the online implementation for power ascension from core power of 3411 MWt to the mini uprated power level which represents an increase of 1% steam flow through the high pressure turbine from the former 100% rated thermal power. The Uprate power ascension procedures do not direct configuration control activities, program impact, procedure impact, process computer impact and training impact reviews. The power uprate master DCP9900597 directs these activities for Unit 1 and DCP 900598 for Unit 2. Revision 1 to SPP 01-002 has removed the requirement to perform the DCP 9900518 (Main feedwater pump flow/speed control and turbine driven feedwater pump electro-hydraulic controller calibration) for power uprate. The master DCP 9900598 controlling power uprate activities was also revised along with 50.59 applicability being done to reflect changes.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the power ascension will not result in any equipment important to safety to malfunction do to being operated, shutdown, or isolated such as to impair its ability to perform its safety related mitigating functions. The power ascension from the current license power level to the power uprate level does not impair any safety-related component performance during accident or transient conditions. Therefore, the proposed activity does not increase the consequences of any malfunction of equipment during power ascension.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the configuration, operation and accident response of the Braidwood Unit 1 and 2 components are unchanged by operation during power ascension. Analyses of transient events as a result of uprate have confirmed that no equipment malfunction results in new sequence of events that could lead to a new or different accident scenario. The effect of operation at the new uprated power conditions was evaluated. No operating mode, equipment lineup, or equipment failure was identified that could result in a new or different accident or transient. The proposed activity does not result in any equipment to operate outside of the power uprate analyses assumptions or equipment design conditions that could result in a new or different malfunction or failure than was analyzed for power uprate.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the proposed power uprate ascension procedures will not reduce the margin of safety as defined in the Technical Specification Basis as amended by NRC's Safety Evaluation Report.

50.59 REVIEW COVERSHEET FORM

BRW-E-2001-338

ıtion: Braidwood

Activity/Document Number: DCP 9900675 (U1). 9900684 (U2): DRP# 9-060 **Revision Number:** 0

Title: Letdown Booster pump modification

Description of Activity:

This DCP provides the design information necessary to install a Residual Heat Removal (RHR) letdown flow booster pump system to each unit in order to increase reactor coolant letdown flow. It is desired to operate the booster pump system during Modes 5 or 6 with reactor coolant temperature less than or equal to 140°F. During normal plant operation, the system will be isolated from RH on the suction side by two, locked closed, manual gate valves and a check valve and manual globe valve on the discharge side.

Presently, during shutdown conditions, reactor coolant letdown flow is provided by the RHR system. The flow rate in the design condition is limited to 120 gpm. With the booster pump in operation, the flow rate is expected to produce a minimum of 150 gpm. and a maximum of 180 gpm. This increased flow rate will reduce the required time to clean up the reactor coolant to acceptable activity levels following the crud burst. This in turn will reduce critical path outage time. The booster pump is intended to be operated following a crud-burst (forced oxidation: normally performed in advance of a refueling outage to lower dose levels from crud contained in the reactor coolant). Operation of the booster pump will occur between the following two extremes: 1) with the RCS water solid and a maximum pressurizer cover-pressure of 5 psig; and 2) at mid loop RCS hot leg level and 0 psig RCS pressure. The maximum RCS temperature will be 140°F.

Westinghouse has confirmed that a flowrate of 180 gpm is acceptable through all CV letdown line components (structural adequacy of the letdown line components has been verified up to 180 gpm). However, they do note that flowrates in excess of 120 gpm may result in mixed bed demineralizer "channeling" - a condition that may inhibit the demineralizer from removing soluble contaminants from the reactor coolant.

The letdown flow booster pump takes suction from the A RHR pump suction line RH0lCA-16", and discharges to the reheat heat exchanger outlet line CV80A-3" at a point between manual valve CV7038 and letdown reheat heat exchanger flow control valve ˜V381A (this flow control valve will be maintained closed since it now provides isolation for the booster pump). Line CV80A-3" connects to the normal letdown line at the same point as the RH tie-in. The flow path then from this point to the VCT remains unchanged by the proposed activity.

Booster pump operation can occur with or without the operation of the A RH pump. Adequate NPSH will be available to the booster pump with the RH pump in operation and likewise, adequate NPSH will be available to the A RH pump with the booster pump in operation. The A RH pump can operate and perform its shutdown cooling function while the booster pump is operating. Refer to calculation BRW-01-0138-M, Rev. 0 for details/limitations. This calculation also demonstrates that the existing letdown line relief valve provides an adequate level of over pressure protection with the booster pump. The elevated pressures the booster pump must develop to produce a 180 gpm letdown flow rate are not in excess of the relief valve set-point or the design pressure of letdown piping and its components.

The booster pump is non-safety related. The booster pump suction and discharge lines are classified as ASME Section III, Class 3. The pump is seismically supported.

The booster pump is located in the letdown chiller heat exchanger room on elevation 346' of the Auxiliary Building. The HX in this room is no longer being used. There is no equipment important to safety located in this room. The booster pump suction pipe is routed to the adjacent 1A (2A) Containment Spray (CS) pump room to tie into the RHR pump suction line 1/2RH01CA-16". The booster pump suction line (approximately 25') is provided with two manual valves located in the A CS pump room to isolate the RHR safety Class 2 piping from the non-safety booster pump The pump discharge line (approximately 26') is routed to the adjacent CV-BR valve room (valve aisle #1) to tie into the discharge pipe of the reheat heat exchanger. The existing check valve CV7039, manual valve CV7038, and AOV CV381A, located in the valve room, are used to isolate the Chemical Volume and Control (CV) safety class piping from the non-safety related pump.

Power for the 75 HP booster pump motor will be taken from the BR chiller compressor starter cabinet outside the L/D chiller HX room. The BR chiller compressor control switch and its associated control circuitry will be modified to operate the booster pump. The Letdown Booster Pump is fed from a non-safety related power source. The pump will only be operated with the unit shutdown, thus, operation of the pump will not occur concurrently with the most limiting design basis event for the Auxiliary Power System. Adequate

BRW-E-2001-338

Station: Braidwood

Activity/Document Number: DCP 9900675 (U1). 9900684 (U2): DRP# 9-060 **Revision Number:** 0

Title: Letdown Booster pump modification

protection is provided to the circuit to ensure that any electrical failure of the pump or control circuit will not propagate and adversely impact the safety class 1E Auxiliary Power System's ability to perform its intended functions.

The booster pump system may be installed while the associated unit is operating The connection of the booster pump suction line to the 16" RH header will require an ECCS LCOAR since the header must be drained to make the connection. The booster pump discharge piping can be isolated from the CVCS when the connection is made (no CVCS impact). Testing and flushing activities associated with the booster pump can be performed with no impact on the plant by closing the discharge and suction valves.

SSCs affected by the proposed activity include the BTRS, RHRS, Train A, and the CVCS. As described in the UFSAR, section 9.3.4.1.2.4, the BTRS is no longer used.

Reason for Activity:

The activity is necessary to shorten the outage time required to cleanup the reactor coolant following a crud-burst The crud-burst is necessary to lower the dose levels associated with the reactor coolant - thereby reducing the personnel exposure. The mixed bed demineralizer(s) removes the soluble contaminants from the reactor coolant. By increasing the flowrate through the demineralizer, the time required to reduce the dose level of the reactor coolant to some acceptable value can be reduced. EXELON has estimated that this increase in flow rate is expected to reduce the clean-up time by approximately 10 hours.

Effect of Activity:

The proposed activity results in an increase in the letdown line flowrate from 120 gpm to a maximum of 180 gpm. The activity will affect the BTRS chiller compressor and its associated control circuitry. Power originally provided to the chiller compressor will be used to power the 75 HP booster pump motor. The hand switch, originally used to control the chiller compressor, will be used to control the booster pump. BR piping in the letdown chiller HX room may be relocated or cut and capped. The BTRS, as it was designed originally, will be rendered inoperable by the proposed activity.

The higher letdown line flow rate (maximum of 180 gpm) resulting from the proposed activity has been evaluated by Westinghouse and deemed acceptable for the letdown piping and associated components/equipment.

Summary of Conclusion for the Activities 50.59 Review:

This activity does 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, there is no possibility of creating an accident of a 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 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-2001-338	**Rev.**	0
[ ]	**50.59 Validation**	**50.59 Validation No.**		**Rev.**	

Tracking Number: BRW-V-2001-342

UFSAR REVISION
UFSAR Draft Revision Package 9-019

DESCRIPTION

This evaluation was performed to evaluate an update to the UFSAR. This activity updated the Licensing Basis to clarify the administrative duties of the Shift Manager. References to non-docketed studies performed in 1977 and 1979 are being deleted and replaced with commitments pertaining to the administrative duties of the Shift Manager.

SAFETY EVALUATION SUMMARY

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report was not increased because this change is administrative in nature. No physical change to the facility were performed. No significant change to normal operations occurred.

2. The possibility for an accident or malfunction of a different type other than any evaluated previously in the safety analysis report was not created because there were no physical change to the plant. Therefore, all previous analyses remain valid. No new accidents were possible.

3. The margin of safety, as defined in the basis for any Technical Specification, was not reduced because there was no physical change to the plant. Therefore, the basis of the Technical Specifications remained unchanged and the margin of safety was unaffected.

PROCEDURE REVISION

BwEP ES-0.1

DESCRIPTION

The purpose of this revision was to change the PRT high level alarm set point from 80% to 88% per DCP 99000503 and SSCR 00-090.

SAFETY EVALUATION SUMMARY

The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because components will operate in a manner consistent with the current practice. The change in alarm set point has been evaluated to be acceptable as part of the change process (DCP 99000503).

50.59 REVIEW COVERSHEET FORM

BRW-E-2001-366

Station: Braidwood

Activity/Document Number: Design Changes / Unit 1 EC/DCP 9900678 & Unit 2 EC/DCP 9900680 **Revision Number:** 0

Title: Revenue Metering for Main Power Transformers (MPT)

Description of Activity:

These DCPs will install outdoor free-standing combined current and voltage (potential) transformers (CT/PT) in the station Switchyard for metering station electrical power output. These combined CT/PT transformers will be installed in-line on each phase of the 345 KV lines from the Main Power Transformers (MPTs) to the Switchyard. These DCPs implement changes in the switchyard, and impact the electrical conductors from the MPTs to the Switchyard which conduct the electrical power output of the station main Generators to the Switchyard (grid). These changes affect SSCs associated with the Switchyard (SY) system.

Reason for Activity:

The CT/PT units are being installed in the 345 KV lines from the MPTs to the switchyard in order to meter power output of the Generators to the transmission grid. The ComEd Interconnection Agreement, FERC open access transmission rule and ICC regulations require metering at the point of interconnection (high side of transformer) to the ComEd transmission system.

Effect of Activity:

The affect of these changes will be to physically install and electrically connect combined CT/PT units in-line on the 345 KV lines from the MPT's to the switchyard. There will be no effect on plant equipment or plant operation when the CT/PTs function as designed. However, failure of a CT/PT could result in a loss of external electrical load accident and subsequent generator trip event, turbine trip accident, and reactor trip event. However, the effect and consequences of this failure are no different than those of other failures which cause the loss of this path to the switchyard and are bounded by existing UFSAR analyses. Loss of External Electrical Load (UFSAR 15.2.2) and Turbine Trip (UFSAR 15.2.3) are Category II faults, which at worst, result in a reactor trip with the plant capable of returning to operation.

Due to the complexity of the CT/PTs compared to the 345 KV line, the likelihood of a malfunction has increased and subsequently, the likelihood of initiating an accident. However, based on CT/PT design and construction compliance with applicable GDC, UFSAR requirements and commitments, and applicable industry standards, Engineering has determined that there will not be more than a minimal increase in either the likelihood of a malfunction or the initiation (frequency) of an accident.

Summary of Conclusion for the Activities 50.59 Review:

Installation of the CT/PTs in-line with the electrical conductors from the MPTs to the switchyard are design changes to the plant which affect SSCs that perform design functions described in the UFSAR. The installation of the CT/PT units was considered adverse due to the potential reduction in reliability due to the increase in complexity of the CT/PT compared to the 345 KV conductor, which could potentially increase both the likelihood of a malfunction and the likelihood (frequency) of an accident. Therefore, a 50.59 evaluation was performed.

Malfunction or failure of a CT/PT unit could result in the initiation of a loss of external electrical load and turbine trip accidents. However, CT/PT failure is bounded by existing UFSAR accident analyses for loss of external load (UFSAR 15.2.2) and turbine trip (UFSAR 15.2.3), which determined that neither accident will prevent safe shutdown of the reactor or adversely affect mitigation of an accident or accident consequences. Although a new component susceptible to failure is being installed, the result and consequences of a malfunction or failure is not different than previously analyzed and is therefore bounded by existing analyses. In addition, based on CT/PT design and construction compliance with applicable industry standards, Engineering has determined that there will not be more than a minimal increase in the likelihood of a malfunction or initiation (frequency) of an accident.

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-2001-366**	**Rev.**	0
	50.59 Validation	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

LS-AA-104-1001
01/11/01

ation: Braidwood

Activity/Document Number: BRW-V-2001-432 Revision Number: 0

Title: DRP 9-042 (update B/B UFSAR Section 4.3.2.4 and Table 4.3-2)

Description of Activity:

This activity updates the B/B UFSAR Section 4.3.2.4 and Table 4.3-2. The rod worth uncertainties listed in the above-mentioned UFSAR sections will be revised from 10% to 7%.

Safety Evaluation BRW-E-2001-301 was generated to revise plant procedures (BwVS 500-6) with the new rod worth uncertainties. While the 50.59 discussed a UFSAR change, the change was not documented in the Activity/Document section. Hence, this validation (BRW-V-2001-432) is being generated to complete the UFSAR update.

Reason for Activity:

This activity will make the UFSAR consistent with the Shutdown Margin (SDM) methodology used for Power Uprate. Specifically, the Control Rod Worth uncertainty used in SDM calculations was revised from 10% to 7%. This methodology has been approved by the NRC, is appropriate for the intended application and within the limitations of the SER granted.

fect of Activity:

UFSAR sections 4.3.2.4 and Table 4.3-2 will be consistent with methodology utilized in Power Uprate analysis.

Summary of Conclusion for the Activities 50.59 Review:

This UFSAR update can be completed 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.

Forms Attached: (Check all that apply.)

☐	Applicability Review				
☐	50.59 Screening	50.59 Screening No.		Rev.	
☐	50.59 Evaluation	50.59 Evaluation No.		Rev.	
x	50.59 Validation	50.59 Validation No.	BRW-V-2001-432	Rev.	0

TECHNICAL BASES REVISION

01-003

DESCRIPTION

The purpose of this Technical Bases Revision was to revise the time that the reactor shall be subcritical from> 100 hrs to > 72 hrs with average water temperature of UHS ≤ 100°F for A1R09, before commencing movement of irradiated fuel from the reactor vessel to the Spent Fuel Pool (SFP). Technical Specification Bases Section 3.9.4, "Containment Penetrations," was also revised to reference the new required decay time of> 72 hours. DRP 9-066 will change the UFSAR to reflect the option of the shorter ICDT.

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the proposed change does not increase the failure rate of the refueling equipment or human error. The consequences of the accident are not increased since (1) the calculated dose increase due to the offload time change is more than offset by increased filter efficiencies, and by conservatism in the power level and peaking factor assumed in the analysis and (2) the total does calculated remains below the NRC approved limit of 75 Rem to the thyroid and 25 Rem to the whole body. The radiation monitoring equipment that is required to operate in support of the assumptions in the accident analysis has been found to be qualified for the dose rate due to a Fuel Handling Accident with an In Core Decay Time (ICDT) of 72 hours. The Fuel Handling Building emergency exhaust filter train components, fans, isolation dampers, and instrumentation are not affected by the ICDT change.

   The probability of the loss of spent fuel pool cooling is not increased as a result of reducing the ICDT. In the event of the failure of a spent fuel pool pump or loss of cooling to a spent fuel pool heat exchanger, the second cooling train provides 100% backup capability, thus assuring continued cooling of the spent fuel pit. The ICDT has no bearing on the failure probabilities of the Spent Fuel Pool Cooling System (SFPCS).

   The consequences of a loss of spent fuel pool cooling are not increased as a result of reducing the ICDT to 72 hrs. The additional decay heat input into the SFP due to the earlier core offload time has been evaluated. The increase in heat load of 1 MBTU/hr from a 72 hours ICDT is more than offset by the reduction in background heat load of about 5.4 MIBTU/hr from the current spent fuel pool fuel inventory. Therefore, the maximum fuel pool temperature and the time to boil from an ICDT of 72 hrs for refueling outage A1R09 is bounded by the current design basis analysis for the spent fuel pool. The licensing amendment for the SFP re-rack project has been approved by the NRC via letter addressed to O.D. Kingsley (ComEd) from G.F. Dick (NRR) dated March 1, 2000.

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the proposed change does not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. The proposed change does not affect the capability of the fuel handling equipment. Thus, it is concluded that the proposed change does not create a new or different kind of accident.

   The proposed change only affects the ICDT that the spent fuel assemblies can be moved from the reactor core into the spent fuel pool. The fuel transfer will be controlled by approved Station procedures and there will be no changes to the fuel handling equipment. The fuel pool temperature resulting from the full core fuel transfer has been evaluated to be below the design limits for the SFPCS equipment. Therefore, there is no increase in the possibility of a different type of malfunction of equipment important to safety than any previously evaluated.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because Tech spec 3.3.8 "Fuel Handling Building (FHB) area radiation monitor is to initiate, upon a radiation signal, the FHB ventilation system to ensure that radiation material in the FHB atmosphere are filtered and adsorbed prior to being exhausted to the environment. The area monitors have been evaluated to be able to perform their design function under the radiation field resulting from the FHA of an assembly with an ICDT of 72 hours. Consequently, the proposed change does not involve a reduction in the margin of safety.

   The change in the temperature of the spent fuel pool water was evaluated for the potential increase in reactivity. The design basis criticality analysis was performed assuming a spent fuel pool water temperature of 4°C (39°F), which is well below the spent fuel pool temperature during refueling time. Because the reactivity temperature coefficient in the spent fuel pool is negative, temperatures greater than 4°C will result in a decrease in reactivity. The effect of a dropped fuel assembly on the criticality of the spent fuel pool was also evaluated in the design basis criticality analysis. Reducing the ICDT to 72 hours does not alter the damage caused by the impact of a dropped assembly. Criticality of the spent fuel pool will remain < 0.95. the proposed change does not involve a reduction in the margin of safety.

Safety Evaluation Summary Form

Tracking No. BRW-E-2001-448
Activity No. BwVP 850-22

DESCRIPTION:

Braidwood Power Uprate Project Pre and Post Installation Electrical Output Test.

SAFETY EVALUATION SUMMARY:

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

No activity in BwVP 850-22 testing affects any initial condition, assumptions, or status of equipment and systems described in UFSAR, Table 15.0-7 "Plant Systems and Equipment Available for Transients and Accident Conditions". Therefore, the Braidwood Power Uprate Project Pre and Post Installation Electrical Output Test will not alter radioactive consequences described in UFSAR Chapter 15, and the probability of occurrence of an accident or transient is not increased.

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

During the performance of this activity, the plant can be operated in a safe manner consistent with Technical Specifications with some secondary plant systems temporarily deviating from UFSAR descriptions. However, the impacts are within the design basis, and do not impact the plant safety analysis. As a result, operations and alterations performed under this activity will not create the possibility of an accident or transient of a different type than previously evaluated.

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

During the performance of this activity, the plant can be operated in a safe manner consistent with Technical Specifications with some secondary plant systems temporarily deviating from UFSAR descriptions. The impacts are within the design basis, and do not impact the plant safety analysis. As a result, operations and alterations performed under this activity will not reduce the Margin of Safety.

50.59 REVIEW COVERSHEET FORM

BRW-E-2001-483

Station: Braidwood

Activity/Document Number: SPP-01-003 **Revision Number:** 0

Title: Power Uprate Project - Full Power Ascension Procedure

Description of Activity:

The Full Power Uprate Ascension Procedure directs the on-line implementation of the Braidwood Unit 1 power ascension from the pre-outage Administrative Limits of approximately 97.6% power as measured by Turbine Impulse Pressure for Unit 1 to the uprated power level of 3586.6 MWt. The Full Power Uprate Ascension Procedure identifies the expected operating parameters for power ascension evolution. In addition acceptance limits are specified for ensuring the Plant Parameters are maintained within the power uprate analyses limits. Level 1 parameters will be monitored continuously and power ascension will be stopped if any of these parameters exceed the acceptance limits. Level 2 parameters will be monitored periodically during the ramp and data collected at the Final Plateau.

Reason for Activity:

DCPs 9900597 and 9900598 provide the Engineering controls to ensure that changes to plant design documents, operating procedures and the UFSAR are made consistent with the NRC Safety Evaluation Report (SER) approving Power Uprate. Additionally, the DCPs control the implementation of Power Uprate by ensuring that all required activities (including physical plant changes) are completed prior to proceeding with the uprated licensed core thermal power level of 3586.6 MWt. The control of the implementation of the Power Uprate is performed by the power ascension testing requirements specified in the Power Uprate Project - Full Power Ascension Procedures SPP-01-003 (Unit 1).

Effect of Activity:

The Full Power Uprate Ascension Procedures will:

- Demonstrate that affected plant parameters and equipment performance remain within acceptable limits as power is increased to the uprated power level of 3586.6 MWt.
- Provide management oversight and control of the activities, including approval of test data to assure safe operation of Braidwood Unit 1 at the uprated power level of 3586.6 MWt.
- Provide instructions on testing and operational maneuvers to be performed as power is increased.
- Provide for the collection of data used to assess equipment performance during power escalation, and to confirm acceptability for continued testing at the uprated power level of 3586.6 MWt.
- Ensure that plant radiation surveys for selected areas have been completed following power increase and that radiation survey maps and RWP's have been updated as required to control radiation exposure of station personnel.

Summary of Conclusion for the Activities 50.59 Review:

The 50.59 Evaluation demonstrates that the Full Power Uprate Ascension procedure implementation does not require 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.

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-2001-483	**Rev.**	0
[ ]	**50.59 Validation**	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

BRW-V-2001-545

tion: Braidwood

Activity/Document Number: BwAR 1-10-A7 **Revision Number:** 12

Title: ROD DEV POWER RNG TILT Annunciator Response

Description of Activity:

Change setpoint in step A.3.a and D.2.c from 228 steps to 225 steps. The RCCS park position is being changed during A1R09 from 228 steps to 225 steps for all Control and Shutdown banks to be utilized by Braidwood Unit 1 during fuel cycle 10.

Reason for Activity:

This change is being instituted to reduce control rod wear at the guide cards and extend the life of the control rods.

Effect of Activity:

All of these rod positions reside above the active fuel stack and therefore have no neutronic impact. This change will not affect shutdown margin.

Summary of Conclusion for the Activities 50.59 Review:

The previous activity describes all the Braidwood Unit 1 Cycle 10 core reload changes, one of which was to change the RCCA park position from 228 steps to 225 steps. There are no differences between the proposed activity and the previous activity. The previous activity justified changing the RCCA park position and the proposed activity simply incorporates the new position into plant procedures.

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

Forms Attached: (Check all that apply.)

☐	Applicability Review				
☐	50.59 Screening	50.59 Screening No.		Rev.	
☐	50.59 Evaluation	50.59 Evaluation No.		Rev.	
X	50.59 Validation	50.59 Validation No.	BRW-V-2001-545	Rev.	0

50.59 REVIEW COVERSHEET FORM

BRW-V-2001-573

Station: Braidwood

Activity/Document Number: 0BwOA SECURITY-1 **Revision Number:** 0

Title: Security Threat

Description of Activity:

Write procedure to describe actions necessary to mitigate an actual hostile force intrusion into the protected or vital areas, this includes tripping both Units and placing them in a cooldown. The procedure will also provide contingency actions implemented during an elevated security condition that would better prepare the station to handle an actual attack if it were to occur. During an actual plant attack both Units will be tripped and placed in cooldown per applicable procedures. The Main Control Room (MCR) ventilation will be placed in the emergency mode with the recirculation Charcoal absorbers in service. If accessible, the licensed operators will man the Remote Shutdown panels for each Unit. In the event that this procedure is entered as a result of a heightened security level, actions will include the following: maximizing inventory of make-up water sources, verifying fire protection header is pressurized, returning systems to full operational status, verifying adequate staffing levels and other administrative functions that could enhance the station readiness.

Reason for Activity:

The NRC issued a set of recommendations to the industry due to increased security threat imposed by recent terrorist activity. This procedure incorporates these recommendations

Effect of Activity:

Increase safety of plant equipment and personnel during a postulated terrorist attack. This activity provides a ready source of information for the plant operators during such an event.

Summary of Conclusion for the Activities 50.59 Review:

This procedure provides a preplanned strategy that is essentially administrative in nature. Any actual directions given for equipment manipulation reference the applicable procedure(s) for execution None of the directed tasks affect the way an SSE is designed to respond during an accident condition. The strategies employed by this procedure contain no FSAR direction and therefore cannot affect the FSAR. This procedure utilizes other approved procedures to accomplish specific tasks so that no new failure mode not previously addressed in the FSAR will be created. Due to these factors, this screening provides sufficient control of this activity precluding the need for a full 10CFR50.59 evaluation.

Attachments:

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

Forms Attached: (Check all that apply.)

[ ]	**Applicability Review**				
[ ]	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
[ ]	**50.59 Evaluation**	**50.59 Evaluation No.**		**Rev.**	
X	**50.59 Validation**	**50.59 Validation No.**	BRW-V-2001-573	**Rev.**	0

50.59 REVIEW COVERSHEET FORM

BRW-V-2001-576

Station: Braidwood (Station 20)

Activity/Document Number: BwAR 1-10-A7 **Revision Number:** 0

Title: ROD DEV POWER RNG TILT

Description of Activity:

The annunciator response has been revised as a result of the change in RCCA park position for 228 steps to 225 steps. The alarm is computer generated and is set to annunciate when the shutdown bank position reaches < 222 steps based on DRPI indication after the shutdown bank had been > 222 steps also based on DRPI. The computer points that trigger the alarm (K0016 and K0017) will be set at a DRPI indication of 219 steps to ensure the alarm is triggered when the shutdown bank is < 222 steps. The alarm setpoint is within the 3 step accuracy of DRPI.

There have also been changes made to the alarm based on bank demand position. The alarm response has been changed so that it will annunciate when any bank is > ARO (currently 225 steps) position or Control Bank D is > 223 steps. The previous revision had the alarm setpoints of 225 steps for both of these conditions. These changes are due to the new RCCA park position of 225 steps.

Information has also been added to specify that 1BwOS NR-2 and NR-3 be performed only when reactor power is greater than 50% and that 1BwOSR 3.2.4.1 is to be performed as required by TRM TLCO 3.3.h.

Reason for Activity:

The change in the RCCA park position is being made to reduce control rod wear at the guide cards. The changes made to the annunciator response relating to rod position are a result of the change in RCCA park position.

The changes made to the annunciator response to describe when the associated surveillances are to be performed should ensure the procedures are performed when required

Effect of Activity:

All of the rods reside above the active fuel area and therefore the change in RCCA park position has no neutronic affect. This change will not affect shutdown margin.

The changes made that provide the information for surveillance performance are editorial in nature since there is no affect on the performance, intent, or outcome of the annunciator response or the surveillances listed.

Summary of Conclusion for the Activities 50.59 Review:

The previous activity describes all of the Braidwood Unit 1 Cycle 10 core reload changes, one of which was to change the RCCA park position from 228 steps to 225 steps. There is no difference between the proposed activity and the previous activity. The previous activity justified changing the RCCA park position and the proposed activity simply incorporates the new park position into the plant procedures.

The changes made to provide the additional information for surveillance performance are editorial in nature since they do not affect the performance, intent, or outcome of the annunciator response or surveillances referenced. Editorial changes do not constitute a change to the procedure.

Forms Attached: (Check all that apply.)

[ ]	Applicability Review				
[ ]	50.59 Screening	50.59 Screening No.		Rev.	
[ ]	50.59 Evaluation	50.59 Evaluation No.		Rev.	
X	50.59 Validation	50.59 Validation No.	BRW-V-2001-576	Rev.	0

50.59 REVIEW COVERSHEET FORM

ıtion: Braidwood
Activity/Document Number: ITS Bases Change Request 01-016 **Revision Number:** 0
Title: Technical Specification B3.4.15 revision to incorporate containment sump level monıtors PC002 and PC003 as an alternate means to monitor leakage inside containment

Description of Activity:
The proposed activity revises Technical Specification Bases B3 4 15 such that the contaınment floor drain sump flow monitor RF008 is normally utilized to fulfill the containment sump monitoring requirement per Technıcal Specification 3.4 15, but allows containment sump level monitors PC002 or PC003 to be used in place of the RF008 monitor to perform the same monitoring function by comparing the change in sump level over a period of time. More specifically, this activity updates the Technical Specificatıon Bases B3.4.15 BACKGROUND to state "The containment sump, used to collect unidentified LEAKAGE, is ınstrumented to identify leakages of 1.0 gpm within one hour. This sensitivıty is acceptable for detecting increases in unidentıfied LEAKAGE " The Technıcal Specification Bases B3.4.15 LCO has been modified to state, "The containment floor drain sump flow monıtor (RF008) and the reactor cavity sump flow monitor (RF010) are normally utilized to fulfill the containment sump monitor requirement Alarms are provided to alert the operator of leakages of 1.0 gpm. When the alarm function is not capable of detecting 1.0 gpm of unidentified LEAKAGE within one hour, the contaınment floor drain sump flow indıcation may be periodically monitored to ensure capabilıty of detecting 1.0 gpm of unidentified LEAKAGE within one hour. In lieu of the containment floor drain sump flow monıtor (RF008), either containment sump level monitor (PC002 or PC003) can be used by monitoring a change in sump level over a perıod of tıme in such a manner as to ensure the capability of detecting 1.0 gpm unidentıfied LEAKAGE wıthin one hour."

Reason for Activity:
The proposed activity provides an alternate method to identify RCS leakage in the event containment floor drain sump flow monıtor RF008 becomes inoperable, thereby, preventing unnecessary plant transients when acceptable alternate methods are available.

Effect of Activity:
The bases change provides for an acceptable alternate means to monitor leakage inside contaınment in the event the containment floor ·ain sump flow monitor (RF008) becomes inoperable. This would prevent any unnecessary administrative plant transients required by ırrent Technical Specification requirements due to the RF008 when an acceptable alternate means of monitoring leakage is available.

Summary of Conclusion for the Activities 50.59 Review:
Considering an accident previously evaluated in the UFSAR, such as a gross failure of the RCPB, the proposed activity does not result in an increase ın the frequency of occurrence of such an evaluated event since the leakage detection monitoring ınstrumentation is not an initiatör of any accident and no new faılure modes are introduced With the properties and attributes of the SSCs important to safety remainıng unchanged, the actıvity does not introduce an increase in the likelıhood of a malfunctıon of an SSC important to safety previously evaluated ın the UFSAR. The consequences of an accident, and in particular those accidents involving the breach of the RCPB such as a LOCA, as previously evaluated in the UFSAR remain unchanged sınce the proposed activity does not fault any SSC mıtigatıve features used to lessen the consequences of the design basis accıdent. Since the proposed activity can preserve the containment sump monitoring function in the event RF008 loses its detection function, and the proposed change does not ıntroduce ınitiators of any new malfunctıons of an SSC important to safety previously evaluated in the UFSAR, the consequences of such a malfunction are unchanged by the proposed activıty

The proposed actıvıty does not create the possibility of an accident of a different type than prevıously evaluated in the UFSAR because the activity, being both passive and data gathering in nature, is not an inıtiator of any accident includıng those involving the RCPB, and no new faılure modes are introduced in performing such an activity The leakage detection monitoring subsystems associated with this change are non-intrusive to any pressurized retaining process fluid boundary. The proposed activity does not alter the physical RCPB and does not alter or control mechanisms that may change the RCPB parameters such as RCPB stresses, RCS thermal heat load, or RCS pressure The acceptance limits for fission product barriers is preserved Incorporatıng an alternate method to monitor leakage to the containment floor drain is currently supported by the UFSAR. The requirement to detect leak before break ıs still maintained

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-2001-650	Rev.	0
☐	50.59 Validation	50.59 Validation No.		Rev.	

50.59 REVIEW COVERSHEET FORM

6G-02-0003

Station: Byron / Braidwood

Activity/Document Number: DRP # 9-078

Revision Number 0 Date : 01/29/02

Title: Revise UFSAR section 6.3.5.4 using EQ containment floor water level instrumentations for re-circulation switchover from RWST

Description of Activity:

Revise UFSAR section 6.3.5.4 to clarify that EQ, safety related level instrumentation for containment floor water level would be used in the EOPs.

Reason for Activity:

Existing containment (CNMT) sump level instrumentation (LS-940A & LS-941A) is not EQ and Non safety related. This was identified while the review of Emergency Operating Procedures (EOPs). In an effort to compare practices, Braidwood procedures were also reviewed. Subsequently Braidwood deleted references to the CNMT sump level indication from the affected EOPs and modified the steps to address the use of CNMT floor water level instrumentation. CNMT water level instrumentation (LT-PC006 & LT-PC007 loops) are safety related, EQ, seismically mounted and Reg. Guide 1.97 type-1. These instruments provide more reliable analog indication in the control room. However, this change affects UFSAR section 6.3.5.4 and needs revision. DRP #9-078 is prepared for this purpose.

Effect of Activity: Affected:

EOPs will be revised. Since procedures will be revised and IE and Non-1E both equipment provide indication in the control room Operator's may require training on the revised EOP procedures. UFSAR section 6.3.5.4 will be revised per DRP 9-078.

Summary of Conclusion for the Activities 50.59 Review:

Level transmitters LT-PC006 & LT-PC007 loops were considered additional indications to LS-940A & LS-941A for CNMT floor/sump water level (LVL). Procedures were using both (floor and sump LVL instruments) as information for re-circulation switch over from the Refueling Water Storage Tank (RWST) to CNMT floor/sump in addition to the indication of RWST Low Low Level (Refer DRP 6-072). Therefore clarification of Non-1E indication and removal of Non-1E LS-940A & LS-941A from the EOPs for reliability, will not degrade the safety function performed by LT-PC006 and LT-PC007. There is no change in the functions performed by LT-PC006 & LT-PC007. Engineering calculation for the ECCS screen backlog and weir flow (BRW-98-0100-M/ BYR98-030) identified that a minimum 9" water level above the Containment floor (377') is required to ensure adequate weir flow (both RH pumps in operation) at the time of switchover from RWST to containment sump. Per the revised uncertainty calculation #NED-1-EIC-0082, the adverse instrument (LT-PC006 & 007) uncertainty is +/- 10". This identifies that the normal indicated floor level would be 13" which is the EOP set point. Per Braidwood letter ED-BRW-98-0347 (Dated 4/16/98) normal expected minimum floor level is 25". Based on this information the EOP set point of 13" is acceptable and will provide timely switchover information to operator per the EOPs. This DRP will revise the UFSAR to plant condition and will initiate EOP revision.

The 50.59 evaluation has concluded that the proposed activity does not result in an increase in the frequency of occurrence of any accident, does not result in a malfunction and/or consequences of malfunction affecting any safety related and/or related to safety SSC, does not increase the possibility of a new accident or result in a DBLFPB or the departure from the method of evaluation. Therefore the proposed activity may be implemented and the affected EOPs will be revised without obtaining a License Amendment.

No NRC notification is required for this change.

For this 50.59 the following sections of the UFSAR, Tech Specs and Engineer documents were reviewed

UFSAR chapter-6, section 6.3- Emergency Core Cooling Systems (ECCS).
UFSAR chapter-6, section 6.2- Containment Systems.
UFSAR chapter-15, section 15.6- Decrease in Reactor Coolant Inventory.
UFSAR chapter-6, section 6.1.3.3- Loss Of Coolant Accident. (LOCA)
UFSAR chapter-6, section 6.3.5.4- Level Indication (Containment Recirc. Sump Level).
Tech. Spec. 3 /4.3.2- ESFAS Instrumentation.
Tech. Spec. 3 /4.5.5- ECCS refueling water storage tank.
Engineering Calc. SITH-1- RWST level setpoints rev.4
Safety Evaluation T1-96-0080 1 /2BEP ES-1.3 procedure change.
DRP #6-072 RWST level set point calc. revision affecting UFSAR.

6G-02-0003

Station: Byron / Braidwood

Page 1 of 2

Activity/Document Number: DRP # 9-078

Revision Number 0 Date : 01/29/02

Title: Revise UFSAR section 6.3.5.4 using EQ containment floor water level instrumentations for re-circulation switchover from RWST

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).

x	**Applicability Review**				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
x	**50.59 Evaluation**	**50.59 Evaluation No.**	6G-02-0003	**Rev.**	0
	50.59 Validation	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

Station:Braidwood

Activity/Document Number: **BwOP RH-6** **Revision Number:25**

Title: **Placing the RH System in Shutdown Cooling**

Description of Activity:

Operating Procedure BwOP RH-6 is being revised by creating two different steps for RH system startup based on existing RCS temperatures. For RCS temperatures at or below 260 degrees, the RH system startup remains unchanged from previous versions of this procedure. For RCS temperatures above 260 degrees (up to 350 degrees), a new sequence of steps is provided to insure RH system warm-up rates are not exceeded. The primary difference will be the direction to de-energize the pump recirculation valve in the open position to prevent uncontrolled temperature transients. Furthermore, additional Precautions and Limitations and Actions are provided to clarify the concern associated with starting the RH system with RCS temperatures above 260 degrees.

Reason for Activity:

This revision is required to provide the necessary guidance to the Main Control Room Operators to insure the recommendations described in Westinghouse Technical Bulletin ESBU-TB-96-03, RHR Pump Operating Recommendations, are addressed. This will insure the integrity of the RHR system and eliminate potential problems caused by rapid temperature changes.

Effect of Activity:

This revision will provide the necessary guidance to the operators to prevent potential damage to the RH pump caused by excessive system heat-up rates.

Summary of Conclusion for the Activities 50.59 Review:

This procedure revision may be implemented without NRC approval since the operation of the RH system is not an initiator to any accident and will not alter the consequences of any analyzed accident, or create the possibility of a different type of accident, when operated in accordance with this revised procedure. All potentially affected SSCs, and the potential failure modes, are bounded by the existing Failure Modes and Effects Analyses, hence, the consequences of any failure of an SSC remain unchanged. The potential for increased failure rates of the circuit breaker associated with the recirculation valve is extremely low based on usage and existing preventative maintenance programs. This revision does not affect the operation of the required ECCS train therefore there is no effect on any Design Basis for Fission Product Barrier. Finally, this procedure does not involve a method of evaluation, hence, there is no departure from a method of evaluation described in the UFSAR.

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-2002-010**	**Rev.**	**0**
☐	**50.59 Validation**	**50.59 Validation No.**		**Rev.**	

UFSAR REVISION
DRP 9-076

DESCRIPTION

The purpose of this UFSAR revision is to Add a description of the Fuel Assemblies' debris filter bottom nozzle to the UFSAR discussion of compliance to Regulatory Guide 1.82, "Sumps for Emergency Cooling and Containment Spray systems". A protective grid has been added to the bottom nozzle of each fuel assembly to provide a zone, below the active fuel, where debris can be trapped. The grid is arranged to sit on top of bottom fuel nozzle so that the intersection of the grid straps is over the bottom nozzle (See attached Figure 1). This reduces the possibility of fuel rod damage due to debris-induced fuel rod fretting. However, the coolant flow holes through these nozzles are smaller than the size of the inner screen in the containment Emergency Recirculation sumps. The size of the containment emergency sump inner screen square openings is 0.1875 inches (resulting in a diagonal dimension of 0.265 inches) while the size of the opening at the bottom of the fuel assemblies is 0.07 inches (See attached Figure 1). This activity will not have any impact on the operation of any plant systems. The new design fuel assemblies have already been used at both Byron and Braidwood as described in Chapter 4 of the UFSAR (Refer to PIF #A1999-02706).

SAFETY EVALUATION SUMMARY

1. The probability of an occurrence, or the consequences of an accident, or a malfunction of equipment important to safety, as previously evaluated in the UFSAR, is not increased because the evaluation of this scenario and concluded that the presence of the small flow channels at the bottom of the fuel assemblies does not pose a significant risk to core cooling under accident conditions. Although some blockage may occur, the minimum flow into the core as a result of the blockage is well in excess of that required to remove decay heat at the time that coolant is drawn from the containment sumps. The technical evaluation addressed a hot leg RCS break scenario; this is the limiting case. If the break is in the cold leg of the RCS, the flow rate into the core is reduced due to the loss of safety injection flow through the break, and resistance to flow downstream of the core. If the break is in a hot leg, all safety injection flow will enter the core. This higher flow results in more suspended particles reaching the bottom of the core. The evaluation of this issue assumes that flow blockage does occur. This is a conservative assumption. In fact, debris particles that enter the ECCS recirculation flowpath would be further fragmented by the pumps that take suction from the containment recirculation sump (RH), and/or by the pumps served by the RH system during this phase (CV and SI).

2. The possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR is not created because the likelihood of a malfunction of an SSC important to safety, as the change in the bottom grid at each assembly does not increase the frequency of failure of the fuel or any other plant component.

3. The margin of safety, as defined in the Bases of the Technical Specifications, is not reduced because the potential impact on core cooling has been evaluated; adequate Emergency Core Cooling System (ECCS) flows are maintained. Since adequate core cooling is maintained, the consequences (dose) from a Loss of Coolant Accident or a malfunction of an SSC important to safety are not increased. There are no design basis limits for a fission product barrier that are exceeded or altered by the implementation of the modified fuel design.

50.59 REVIEW COVERSHEET FORM

Station: Braidwood Units 1/2

Activity/Document Number: Change BwVP RM-80-3 procedures for "Checksource Limits" reference for the main steam line radiation monitors (1/2AR22/23) radiation monitors . Channel Item 15 "Checksource Limits" is being changed back to the original vendor recommendation counts of 500 counts for all detectors associated with these radiation monitors. See below list of all procedures that are affected by this change.

Revision Number: See below list of procedures

Title: "Checksource Limits" channel item 15 is changing for the main steam line radiation monitors (1/2AR22/23). See below list of all procedures that are affected by this change.

Description of Activity:
The following procedures are included in this change

1. 1BwVP RM-80-3 1AR22 Appendix 1AR22, Data Base File Sheet, Main Steam Line Monitor Rev 8
2. 1BwVP RM-80-3 1AR23 Appendix 1AR23, Data Base File Sheet, Main Steam Line Monitor Rev 4
3. 2BwVP RM-80-3 2AR22 Appendix 2AR22, Data Base File Sheet, Main Steam Line Monitor Rev 2
4. 2BwVP RM-80-3 2AR23 Appendix 2AR23, Data Base File Sheet, Main Steam Line Monitor Rev 6

The above listed procedures were revised to change the "Checksource Limits" time associated with the RM-80 database. All radiation monitors that had Channel Item 15 "Checksource Limit" at a higher number than the vendor recommendation were changed. The " Checksource Limit" is the reference counts that the detector must see once every 24 hours to verify that the detector is responding correctly.

Reason for Activity:

The proposed change is associated with every radiation monitor except for the1/2PR27J Steam Jet Air Ejector Radiation Monitors. Braidwood station currently performs a checksource test on every detector once every 24 hours. A checksource is a self-test that either increases current to provide a higher reading to the detector or inserts a source in front of the detector to verify the detector is seeing the correct amount of counts. The amount of counts was determined by the calibration of the radiation monitor. Braidwood station uses procedure BwVS 900-30 "Checksource Reference", this procedure is performed once a calibration is performed or verified on the detector. The procedure allows for five separate checksource test to be performed, the five checksource are then averaged and sixty five percent of the average is then used as the set point for the checksource.

Vendor recommendations for checksource set points are a maximum of 300 counts is used for iodine channels and maximum of 500 counts are used for all other detectors. Values that are currently less than these numbers are valid due to performing the BwVS 900-30 procedure. This change will incorporate having all Checksource Limits at the maximum number or less. Braidwood station went to the BwVS 900-30 methodologically to try and capture checksource failure causes. This procedure takes 65% of total counts and uses that number for the reference point. Going forward will use the smaller number of this procedure or the vendor recommendation. BwVS 900-30 will be changed to incorporate this change along with the above mention procedures that reference the RM-80 database for Channel Item 15 "Checksource Limits".

Effect of Activity:

There is no effect to the system, structure or component maintenance or operation as described in the Technical)ecifications or UFSAR. The detectors are tested once every 24 hours to prove that the detector is responding correctly. By changing this parameter numerous nuisance alarms will be delete from the radiation monitor system. The availability of these radiation monitors will increase due to the threshold of the checksource limit decreasing. The indicator for unplanned Lcoar entry will decrease as will maintenance and health physics activities do to less unavailability time.

Summary of Conclusion for the Activities 50.59 Review:

The 50.59 Screening Form screened these changes as not requiring a 50.59 evaluation.
The subject changes do not require NRC approval. The changes do not conflict with previously approved license documents. No where is the "Checksource Limit" Channel Item 15 setpoint referenced. The UFSAR does require that "routinely during reactor operations, the detector response is observed with a remotely position check source supplied with the monitors". The " Checksource Limits" function is still provided, but the reference point is being changed back to the original vendor recommended numbers. The subject changes can be implemented.

Attachments:

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

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☐	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
☐	**50.59 Evaluation**	**50.59 Evaluation No.**		**Rev.**	
X	**50.59 Validation**	**50.59 Validation No.**	BRW-V-2002-36	**Rev.**	0

50.59 REVIEW COVERSHEET FORM

BRW-V-2002-41

Station:Braidwood

Activity/Document Number: 2BwOA INST-1, 2BwOA ROD-3, 2BwOA PRI-12, 1BwOA PRI-12 **Revision Number:**102, 101, 100, 101

Title: **Nuclear Instrumentation Malfunction, Dropped or Misaligned Rod, Uncontrolled Dilution**

Description of Activity:

2BwOA INST-1 Rev. 102:

STEP NO.	CHANGE DESCRIPTION	REASON FOR CHANGE	EFFECT OF CHANGE
Entry Conditions	Replaced BDPS FLUX DOUBLED alarm with BORON DILUTION ALERT CHANNEL B alarm.	Reflect BDPS modification. SR input is only to Channel B. It does not input to Channel A.	Ensure procedure reflects plant configuration.
1 Note	Revised note to refer to Emergency director instead of Station director.	Incorporate change to Emergency plan terminology.	No effect. Editorial change.
Att C step 5	Removed steps to realign CV pump suction back to VCT upon a failure of Source Range channel. This involves removal of the steps dealing with the old BDPS system. Also reworded new substep d to reflect bypassing of Flux doubling signal.	BDPS modification removes the automatic BDPS function to align CV pump suction from the VCT to the RWST and removes the BDPS block switches. Flux Doubling signal still exists but now feeds an alarm rather than the BDPS actuation signal.	Ensure actions are not directed for equipment which has been removed.
Att. C Step 6c	Added step to block High Flux at Shutdown alarm.	This alarm will annunciate if the SR channel fails high and should be blocked. This change provides consistency with step 5 for Modes 3, 4, and 5 where this step already exists.	Ensure consistency in procedural direction.
Figure 2BwOA INST 1-1.	Remove 'BDPS' from the title	BDPS is no longer associated with source range flux doubling.	Ensure procedure reflects plant configuration.

2BwOA ROD-3 Rev. 101:

Step No.	Change Description	Reason for Change	Effect of change
1 Note	Revised note to refer to Emergency Plan and Emergency director instead of GSEP and Station director.	Incorporate change to Emergency plan terminology.	No effect. Editorial change.
3 RNO	Changed from placing BDPS switches to RESET, to placing Boron Dilution Alert alarm bypass switches in NORMAL.	BDPS modification removes BDPS block switches and replaces them with Boron Dilution Alert alarm bypass switches. The alarms are bypassed with rods withdrawn in Mode 3 and need to be unblocked when reactor startup aborted.	Ensure operator actions are consistent with BDPS modification.

2BwOA PRI-12 Rev. 100

Step No.	Change Description	Reason for Change	Effect of Change
Entry condition B	Replaced BDPS alarms for BDPS flux doubled and BDPS ACTUATED with BORON DILUTION ALERT CH A & B alarms.	BDPS modification replaces these alarms. These alarms are primary indication of an uncontrolled dilution in Modes 3, 4, or 5.	Ensure procedure reflects plant configuration.
1 Note	Revised GSEP note to delete reference to BwZP 200-1 and changed to Emergency Plan and Emergency director instead of GSEP and Station director.	BwZPs have been eliminated. GSEP evaluation still needs to be performed. Emergency Plan terminology has changed.	No effect. Editorial change.
Step 1	Added new Step 1 to check unit in mode 1 or 2. If not in Mode 1 or 2, RNO action is to swap CV pump suction from VCT to RWST.	Ensure analysis assumption of 15 minutes for operator action to swap CV pump suction is satisfied by taking these actions in Step 1 if not already taken.	Ensure required operator actions are performed promptly.
Step 7	Deleted substep to check BDPS ACTUATED CHG SUCT SWITCHOVER alarm not lit.	BDPS modification eliminated the automatic switchover function.	Ensure procedure reflects plant configuration.
Step 7a RNO	Added "IF the VCT is NOT the source of the uncontrolled dilution, THEN"	Do not want to restore suction to the VCT if the VCT is known to be diluted.	Eliminate the possibliity of reinitiating the dilution.

50.59 REVIEW COVERSHEET FORM

BRW-V-2002-41

10	Added Tech Spec 3.3.9	Completeness of T.S. list.	Ensure completeness of T.S. references.

1BwOA PRI-12 Rev. 101:

Step No.	Change Description	Reason for Change	Effect of Change
9d	Removed 2nd closed bullet statement in step 9d that directed transition to 1BwOA PRI-2.	This statement was accidentally left in the existing procedure due to a typing error.	No effect. Editorial change.

Reason for Activity:

See Above tables

Effect of Activity:

See above tables

Summary of Conclusion for the Activities 50.59 Review:

The activity being implemented does not affect any UFSAR described design function. The revisions to incorporate the BDPS modification were previously evaluated in the associated License Submittal. The activity here simply incorporates this modification and the actions associated with it into the appropriate procedures. The BwOA PRI-12 procedure provides direction for diagnosing an inadvertent dilution and borating the RCS in accordance with statements made in the license submittal. These changes have no adverse impact on UFSAR described design functions. Instead, these changes ensure that UFSAR described actions are appropriately performed.

The changes to the GSEP note are editorial. The change to not reinitiate a dilution from the VCT is an enhancement consistent with the original intent of the procedure but not previously stated in the procedure. This activity does not require changes to the Tech Specs or Operating License other than those already described in the associated Licensing Submittals. No Safety Evaluation is required.

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

50.59 REVIEW COVERSHEET FORM

BRW-V-2002-41

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

Forms Attached: (Check all that apply.)

☐	**Applicability Review**				
☐	**50.59 Screening**	**50.59 Screening No.**		**Rev.**	
☐	**50.59 Evaluation**	**50.59 Evaluation No.**		**Rev.**	
☒	**50.59 Validation**	**50.59 Validation No.**	BRW-V-2002-41	**Rev.**	0

50.59 REVIEW COVERSHEET FORM

BRW-E-2002-71

;tion: Braidwood and Byron

Activity/Document Number: DRP 9-081 and TS Bases Change #02-01 **Revision Number:** 0

Title: Revision of UFSAR and TS Bases for Containment Radiation Monitor Sensitivity and Response Time

Description of Activity:

Revise the following UFSAR sections as described below:

- 5.2.5 to eliminate a statement that RG 1.45 only requires one seismically qualified leak detection system and to add sections 5.2.5.2.1 addressing Radiation Monitor sensitivity and response time and 5.2.5.2.2 addressing Leak Before Break considerations.
- Appendix A1.45 to more accurately describe the clarifications/exceptions between the RG and Byron and Braidwood
- 11.5.2.2.10 to provide a reference back to section 5.2.5.2 and
- Table 11.5-1, to revise the containment radiation monitor setpoints to add "or as low as practicable".

Revise TS Bases B 3.4.15 to:

- Delete wording implying that the rad monitors are rapid,
- Provide a discussion of the containment radiation monitor design conditions, and
- State that the rad monitor setpoints will be set as low as practicable.

Reason for Activity:

UFSAR

- Remove an incorrect statement.
- The actual RCS activity levels are much lower than the levels for which the containment radiation monitors were designed in conjunction with Regulatory Guide 1.45.
- Better tie the leak detection requirements of the containment radiation monitors to their monitoring and sampling function

TS Bases

- Remove an incorrect statement. NUREG 1061, Vol. 3 recognizes that rad monitors are only "fair" indicators of leakage
- Improve clarity with respect to the fact that the radiation monitors may not always be capable of detecting 1 gpm leak within one hour, dependent on the actual RCS and containment background activity, and that the setpoints are set as low practicable.

Effect of Activity:

The effect of this activity is to clarify that, although the containment radiation monitors may not always be capable of detecting a 1 gpm leak within 1 hour, their setpoints are maintained as low as practicable given the objective of detecting a leak of 1 gpm within 1 hour and the constraints of actual RCS and containment background activity, and that the numerous RCS leak detection systems, as a whole, meet the intent of RG 1.45.

Summary of Conclusion for the Activities 50.59 Review:

This activity does not make any physical hardware changes to any RCS leakage detection system. The containment radiation monitors are not initiators of any accident, therefore this activity will not result in any change to the frequency of occurrence of any accident evaluated in the UFSAR. The containment radiation monitors will continue to function as originally designed in accordance with RG 1.45, assuming expected RCS activities defined in UFSAR Table 11.1-4. There are no changes to the surveillance requirements of TS 3.4.15; they will continue to be performed in the current manner. The same Limiting Condition for Operation (LCO) Action Requirements of TS 3 4.15 will also continue to be required should either the containment floor drain sump system or the containment atmosphere radioactivity monitor become inoperable. The radiation monitor setpoints will be set as low as practicable, given the objective of ;tecting a 1 gpm leak in one hour and the constraints of actual RCS and containment background activity levels. It is concluded that the

RCS leakage detection system as a whole (as recognized in NUREG 1061. Vol. 3) continues to support the requirements of LBB and there is minimal likelihood of occurrence of a malfunction of an SSC important to safety. Therefore, leakage due to cracks would continue to be identified prior to breakage. Because the crack will still be identified and the plant shutdown prior to breakage, the consequences of an accident are unaffected by this change, nor will it result in any increase in the consequences of a malfunction of an SSC important to safety. These systems perform monitoring functions only. The stability of any crack in the RCS remains acceptable in accordance with WCAP 14559, Rev. 1. Therefore, this change cannot create the possibility of an accident of a different type than any previously evaluated nor create the possibility of a malfunction of any SSC important to safety with a different result than any previously evaluated. This activity does not make any changes to the RCS pressure boundary. The "leakage crack" evaluated in WCAP 14559 Rev. 1 has been shown to be stable for 40 years when subjected to typical transient loads associated with normal, upset and test conditions. Therefore, the RCS pressure boundary and its ability to act as fission product barrier is not impacted by this change, therefore this change does not cause any design basis limit for a fission product barrier to be altered or exceeded. This change does not involve a departure from any method of evaluation described in the UFSAR.

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-2002-71	**Rev.**	0
	50.59 Validation	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

Station: Braidwood Unit 01

Activity /Document Number: ECR 353917 **Revision Number:** 0

Title: Addition of grab handles inside each of the unit 1 Inlet and Outlet Waterbox Upper Manways

Description of Activity:

>Installation of a grab handle inside each of the unit 1 inlet and outlet waterbox upper manways. These waterboxes are located at elevation 401'-0" in the Turbine Building. There are total of four inlets and four outlet waterboxes. A grab handle is to be installed inside each waterbox, just above the upper manway.

Reason for Activity:

> A grab handle is required to assist personnel exiting waterbox through upper manway in the event an emergency evacuation is required. Currently, there is no device installed above the upper manway to facilitate exiting through upper manways. Grab handles are to be installed to resolve safety concerns.

Effect of Activity:

>The addition of garb handles inside each of the unit 1 inlet and outlet waterbox, just above upper manways, will have no affect on plant operations.

ummary of Conclusion for the Activities 50.59 Review:

>Adding a grab handle on the inside of each waterbox just above the upper manway (to resolve safety concerns) has no affect on plant operations, UFSAR, or Technical Specifications. SAR documents are not affected by this change. A 10CFR50.59 screening (BRW-S-2001-243) was performed for adding grab handles inside the unit 2 inlet and outlet waterboxes. Validation of 10CFR50.59 screening (BRW-S-2001-243) has been performed for the scope of adding grab handles inside the unit 1 inlet and outlet waterboxes.

Reference: 10CFR50.59 Validation Number: BRW-V-2002-76, Revision 0 (Attached)

Attachments:

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

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
	50.59 Evaluation	**50.59 Evaluation No.**		**Rev.**	
X	**50.59 Validation**	**50.59 Validation No.**	BRW-V-2002-76	**Rev.**	0

Previous Screening or Evaluation Number: BRW-S-2001-243 **Revision Number:** 0 **Station:** Braidwood

50.59 VALIDATION FORM

50.59 Validation No. BRW-V-2002-76 **Rev.** 0

Previous Screening or Evaluation Number: BRW-S-2001-243 **Revision Number:** 0 **Station:** Braidwood

I. Describe the previous activity (include reference to DCPs, Procedures, etc.)

>Non-power block design change (ER 9904325) was issued for the addition of two more pumps (and associated suction and discharge piping) to the existing four pumps, for the unit 2 Circulating Water (CW) outlet waterbox dewatering system. A grab handle was also installed inside each of the unit 2 inlet and outlet waterboxes, just above the upper manways.

II. Describe any differences between the proposed activity and the previous activity.

>Per ECR 353917, a grab handle is required to be installed inside each of the unit 1 inlet and outlet waterboxes, just above the upper manways. Non- power block design change (ER 9904325 and 10CFR50.59 screening no. BRW-S-2001-243, Rev. 0) was approved for the installation of grab handles in the each of the unit 2 inlet and outlet waterboxes. Sketches issued with the non-power block design change for the installation of grab handle are applicable to both units 1 and 2. Therefore, there are no differences between the previous activity and proposed activity for the scope of adding grab handles.

III. For each difference identified in item II, provide the justification for applying the previous screening or evaluation to the proposed activity.

>There are no differences between the previous activity and proposed activity for the scope of adding grab handles in each of the inlet and outlet waterboxes.

IV. 50.59 Validation (All questions must be YES to apply validation process)

1.	The proposed activity is entirely encompassed by the previous screening or evaluation.	☒ YES	☐ NO
2.	The proposed activity does not extend beyond the plant mode bounds assumed in the previous screening or evaluation.	☒ YES	☐ NO
3.	There are no equipment lineups, temporary alterations, or modifications that invalidate the previous screening or evaluation.	☒ YES	☐ NO
4.	The previous screening or evaluation found the original activity acceptable (i.e., prior NRC approval was not required)	☒ YES	☐ NO

V. Validation Signoffs*:

Validation Preparer: Atul K. Mahadevia (Print name) Sign: ______ (Signature) Date: 02/27/02

Validation Reviewer: Ed Seibert (Print name) Sign: ______ (Signature) Date: 02/27/02

* Use screening or evaluation approval process consistent with document being validated.

50.59 REVIEW COVERSHEET FORM

BRW-V-2002-93

Station: Braidwood

Activity/Document Number: SPP 01-004 **Revision Number:** 0

Title: Braidwood Unit 2 Power Uprate Project Full Power Ascension Procedure

Description of Activity:

This procedure directs Braidwood Unit 2 power ascension from a pre-installation core power administrative limit of~ 95.8% turbine impulse pressure (3436 MW_T) to the full uprated core power level of 3586.6 MW_T. The Uprate power ascension is accomplished as follows:

- Following A2R09, the unit will be ramped to the full uprate power level of 3586.6 MW_T from the pre-installation core power limit of Governor Valves Wide Open (~95.8% turbine impulse pressure).
- After a one-hour soak period at this final power level, plant systems and equipment will be monitored to confirm acceptability for continued operations at the 3586.6 MW_T plateau.

Reason for Activity:

DCP 9900598 provide the Engineering controls to ensure that changes to plant design documents, operating procedures and the UFSAR are made consistent with the NRC Safety Evaluation Report (SER) approving Power Uprate. Additionally, the DCP control the implementation of Power Uprate by ensuring that all required activities (including physical plant changes) are completed prior to proceeding with the uprated licensed core thermal power level of 3586.6 MWt. The control of the implementation of the Power Uprate is performed by the power ascension testing requirements specified in the Power Uprate Project - Full Power Ascension Procedure SPP-0 1-004.

ect of Activity:

- Demonstrates that affected plant parameters and equipment performance will remain within acceptable limits as power is increased to3586.6 MW_T.
- Provides management oversight and control of the activities to assure safe operation of Braidwood Unit 2 at 3586.6 MW_T.
- Provides for the collection of data used to assess equipment performance during power escalation, and to confirm acceptability for continued operation at 3586.6 MW_T.
- Ensures that plant radiation surveys have been completed following power increases and that radiation survey maps and RWPs have been updated as required to control radiation exposure of station personnel.

Summary of Conclusion for the Activities 50.59 Review:

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-2001-243	**Rev.**	0
X	**50.59 Validation**	**50.59 Validation No.**	BRW-V-2002-93	**Rev.**	0

50.59 REVIEW COVERSHEET FORM

BRW-E-2002-109

ation: Braidwood

Activity/Document Number: Engineering Changes 334855/334853/334854 **Revision Number:** 1

Title: Replace Station Air Compressors' Control Panels

Description of Activity:

These changes will install new control panels on the Station Air Compressors (_SA01C). The new control panels are Elliott (OEM) Digital Communication and Control System (DCCS) phase IIC panels (microcomputer controlled), and will replace the existing Electrical/Pneumatic (E/P) analog control panels. The new control panels have the same footprint as the existing panels, and will bolt into the same location on the compressor skid.

Both the existing and new control systems regulate the inlet and unloading valves, monitor operating parameters (temperatures, pressures, etc.), provide alarm and trip functions, sequence the start-up and shutdown of the unit, and govern the different modes of operation. The major difference between the existing control system and the new control system is that the existing control system utilizes discrete electrical and pneumatic devices such as temperature and pressure switches, relays, timing relays, needle valves, etc. to perform control logic functions and control compressor operation, whereas the new control system utilizes a microcomputer and associated software, firmware, and hardware to perform control logic functions and control compressor operation. The software program is stored on EPROM (Read Only) memory chips and can only be changed by changing the EPROMs, which would be controlled by the design change process. The existing control system utilizes instrumentation such as thermocouples, temperature switches, and pressure switches. The new control system utilizes instrumentation such as Resistance Temperature Detectors (RTDs), temperature transmitters, and pressure transmitters. The basic difference in instrumentation is that with the existing instrumentation, each device (e.g., temperature switch, pressure switch) produced a discrete output such as alarm, interlock, trip, etc., whereas the new instrumentation produces 4-20 mA current signals representing the monitored parameters which are then converted to a digital signal and processed by the microcomputer which determines appropriate control system response.

Reason for Activity:

The existing control system components are experiencing increased failure rates due to age degradation and are becoming obsolete. In addition, the existing E/P panel requires numerous adjustments, at numerous locations inside the panel in order to maintain compressor operation within acceptable limits. The new DCCS panel basically requires no adjustment since the microcomputer controls and optimizes compressor operation. Any changes in setpoints, control modes, etc., are made at the operator interface keypad on the front of the DCCS panel.

The intent of these changes is to improve the performance and reliability of the station air compressors. The replacement of the existing control system/panels is an Elliott (OEM) recommended and approved upgrade for these compressors. Elliott field experience with these compressors has revealed that performance and reliability is improved by replacing the existing E/P control system/panels with the new DCCS control system/panels. These changes have been successfully performed on the same type Elliott compressors worldwide. The new control system/panels being installed are also utilized on newer model Elliott compressors.

Effect of Activity:

These changes affect the station air compressors, which are components of the Service Air (SA) system. Neither the design function of the station air compressors or the SA system will be altered by these changes. The compressors interface with the Auxiliary Power (AP), Annunciator (AN), Instrument Air (IA), and Non-essential Service Water (WS) systems. The SA system, AN system, AP system, and the IA system will be affected by these changes. No other plant equipment or systems will be affected. The effect of these design changes on the compressors and the SA system will be to replace the existing station air compressor E/P analog control systems and panels with DCCS (microcomputer controlled) digital control systems and panels. AP system loading will be altered. However, AP system power sources to the panels are sufficient for the new panels and there is no adverse impact on the AP system. No electrical design evaluations, load monitoring systems or calculations require revision. The effect on the AN system will be the consolidation of compressor trouble alarms into one alarm point/window, and the addition of a compressor "auto-start" alarm point/window. The existing IA supply to the panels utilized by the panels for control purposes will only be affected to the extent required to connect the IA lines to the new panel. However, the new control panels have a Panel cooling system, which operates on IA. The new panel cooling system is thermostat controlled and will require an additional 30 scfm per panel from the IA system when operating. Engineering has determined that this additional loading is within the capability of the IA system. The IA/SA system

BRW-E-2002-109

Station: Braidwood

ctivity/Document Number: Engineering Changes 334855/334853/334854 **Revision Number:** 1

Title: Replace Station Air Compressors' Control Panels

interface/relationship will not be altered. The WS system, which is utilized for compressor cooling, will not be affected since cooling requirements will not be changed. There will be no adverse effect on plant or system operation when the new control system functions as intended. Failures of the new control system will have the same end results as failures of the existing control system (i.e., loss of SAC). The intended effect of these changes is to increase the overall performance and reliability of the station air compressors.

Summary of Conclusion for the Activities 50.59 Review:

Although these changes introduce no adverse effects, the installation of the new control system/panels was conservatively treated as adverse due to the analog to digital upgrade. Therefore, a 50.59 evaluation was performed.

While a malfunction or failure of the new control system/panels could result in the loss of the associated SAC, the loss of a SAC is not an accident initiator. Additionally, the SACs are not safety-related, and not credited in the UFSAR for accident mitigation. The loss of SA and IA would eventually result in a Reactor trip. However, the loss of one SAC would not necessarily result in the loss of SA and IA since there are three station air compressors. The transient would depend on additional factors such as the failure of the standby SAC to start. SAC failure is bounded by the existing safety evaluation presented in UFSAR 9.3.1.3 and failure modes effects and accident analyses presented in UFSAR 6.0 and 15.0 which determined that neither the loss of IA or SA (bounding the loss of a SAC) will prevent safe shutdown of the reactor or adversely affect mitigation of an accident or accident consequences. The loss of SA and IA would at worst result in a Reactor trip (a Condition II-fault of moderate frequency) with the plant being capable of returning to operation. This condition II event is not expected to propagate to cause a more serious fault, i.e., Condition III or IV events or result in fuel rod failures or reactor coolant system or secondary system over pressurization.

Although new 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 control system/panel design, 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.

In summary, the conclusion of the evaluation was that installation of the new SAC control system/panels does not require NRC approval. Therefore, the proposed changes may be implemented per applicable procedures.

Attachments:

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

Forms Attached: (Check all that apply.)

	Applicability Review				
	50.59 Screening	**50.59 Screening No.**		**Rev.**	
X	**50.59 Evaluation**	**50.59 Evaluation No.**	BRW-E-2002-109	**Rev.**	1
	50.59 Validation	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

BRW-E-2002-126

ıtion: Braidwood Unit 1&2 / Byron Unit 1 & 2

Activity/Document Number: UFSAR Change Package #DRP 9-075 **Revision Number:** 0

Title: Modeling Assumptions for the Loss of Offsite Power Analysis

Description of Activity:

UFSAR Change Package #DRP 9-075 revises the evaluation methodology for the Loss Of Offsite Power (LOOP) event (UFSAR Section 15.2.6) to incorporate water relief through the Pressurizer Safety Valves. This DRP is applicable to Byron and Braidwood Stations Units 1 and 2.

Reason for Activity:

One of the acceptance criteria for a Condition II event is that an incident of moderate frequency (Condition II) should not generate a more serious plant condition (i.e., Condition III) without other faults occurring independently. The current main acceptance criterion for the LOOP event, as analyzed by Westinghouse, is that the pressurizer does not become water solid. The primary concern with the pressurizer becoming water solid is the possibility of progressing to a more serious event, i.e. a small break LOCA.

The method used by Westinghouse for the existing analysis of a LOOP event assumes that the Chemical and Volume Control System (OVOS) has no net effect on the Reactor Coolant System (RCS) inventory. In effect, charging and letdown are assumed to balance each other with no mass or energy added to or subtracted from the RCS.

In Nuclear Safety Advisory Letter (NSAL-00-013), Westinghouse identified modified CVCS modeling assumptions which could result in a net addition to the ROS water mass, and thus an increased potential for filling the pressurizer. For Byron and Braidwood, the LOOP causes a loss of instrument air. In this case, several valves in the letdown flow path will fail closed, causing isolation of the letdown flow. The failed close valves include the Letdown Regenerative Heat Exchanger isolation valves (1/2CV8389A/B), the letdown line isolation valves (1/2CV459 and 1/2CV460), the letdown line containment isolation valve (1/2CV8152), and the letdown heat exchanger isolation valves (1/2CV8401A/B). As a result of the LOOP, the normal charging path is also isolated. Similar to the letdown line, the loss of instrument air will cause the charging to the Regenerative Heat Exchanger isolation valves (1/2CV8324A-B) in the normal charging line to fail closed. However, the flow path through the RCPs seal injection lines is not isolated; this flowpath results in an inflow to the ROS during a LOOP event.

Westinghouse performed a LOOP analysis taking into consideration the water mass addition to the ROS via the seal injection path. The results of the analysis indicate that the Pressurizer does become water solid and water relief through the Pressurizer Safety Valves will occur. An additional evaluation was performed that concluded the Safety Valves will not be damaged by the water relief (Westinghouse letter LTR-SEE-01-287). The valves will be able to close following lifting and will maintain their design function to limit the pressure of the RCS. The valves may leak, but this event is enveloped by an Inadvertent Opening of a Pressurizer Safety event; this is also a Condition II event. Thus, the criterion that a Condition II event cannot progress to a Condition Ill event without other faults occurring independently is met.

As a result of the above, DRP 9-075 revises the evaluation methodology for a LOOP to incorporate water relief through the Pressurizer Safety Valves.

Effect of Activity:

The CVCS modeling assumption used in the LOOP analysis performed by Westinghouse is consistent with the plant response following a LOOP event. No changes are required to plant components.

The analysis also assumes that plant operators will restore letdown and therefore terminate the net mass addition to the RCS within one hour from event initiation. The existing plant emergency procedures direct the operators to control letdown flow and to stop all charging pumps if pressurizer level is greater than the level specified in the procedure. Additionally, the operators restore instrument air pressure as directed by plant procedure. It is expected that the operators would restore letdown after this action is completed. These actions are part of existing Plant emergency procedures; the one (1) hour period is considered to be conservative and adequate based on operating personnel experience.

BRW-E-2002-126

Station: Braidwood Unit 1&2 / Byron Unit 1 & 2

tivity/Document Number: UFSAR Change Package #DRP 9-075 **Revision Number:** 0

Title: Modeling Assumptions for the Loss of Offsite Power Analysis

Summary of Conclusion for the Activities 50.59 Review:

This activity does use a new methodology for the evaluation of the LOOP event. This activity is not considered a departure from a method of evaluation described in the UFSAR because the new methodology (Water relief through the Pressurizer Safety Valves) has been approved by the NRC for a similar event for Byron and Braidwood Stations as part of the Power Uprate Safety Evaluation Report. In addition, the radiological consequences based on the new methodology remain bounded by the Steamline Break event. 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.

rms 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-2002-126** **6G-02-0007**	**Rev.**	**0**
☐	**50.59 Validation**	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

BRW-E-2002-138

ation: Braidwood

Activity/Document Number: SPP-02-002 **Revision Number:** 0

Title: Unit 2 Main Turbine Roll for Generator Ventilation Testing

Description of Activity:

The SPP will provide the instructions required to perform a main turbine-generator speed increase from zero speed to the 400 and 520 rpm plateaus for a generator ventilation test following the generator replacement during A2R09. Main turbine speed will be controlled using the installed DEH system in the same manner as a turbine roll during unit startup. The difference between the normal unit startup and the turbine roll for the generator ventilation test is that the reactor will not be critical. The plant will be in Mode 3 at NOP/NOT with the steam dumps controlling Reactor Coolant System (RCS) temperature in the steam pressure mode of operation. The steam created from Reactor Coolant Pump (RCP) heat and decay heat that would normally be dumped to the main condenser through the steam dumps will be used to roll the turbine-generator to the test speed plateaus.

Reason for Activity:

The SPP will allow the Unit 2 turbine generator to be rolled to 400 and 520 rpm so that internal generator air flows can be checked. A new generator will be installed in A2R09 and must be checked to ensure that the internal generator blower is not operating in stall conditions and that adequate gas flow is circulated through all sections of the generator.

Effect of Activity:

There is a potential Reactor Coolant System (RCS) cooldown however the SPP procedure will provide the operator guidance to prevent an adverse plant. effect. The SPP will position the turbine so the Generator Replacement Project team can collect the necessary data at the two speed plateaus to ensure proper operation of the generator blower and cooling system. The Shutdown margin will always be maintained within the required Technical Specification values during the performance of the SPP.

Summary of Conclusion for the Activities 50.59 Review:

Neither the MS, DEH, or EF systems will be operated outside the reference bounds of their design bases as described in the UFSAR or inconsistent with the analyses or descriptions in the UFSAR.

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-2002-138	**Rev.**	0
	50.59 Validation	**50.59 Validation No.**		**Rev.**	

50.59 REVIEW COVERSHEET FORM

BRW-E-2002-162

Station: Braidwood Unit 2

Activity/Document Number: Engineering Change /EC336869 **Revision Number:** 0

Title: Remove One Detector from Service on Unit 2 Post Accident Neutron Monitoring (PANM) Channel B

Description of Activity:

This EC will disconnect the input to the A3 Preamplifier at Wide Range amplifier 2NRI3EB for the Channel B PANM. This change removes one detector (A3) from service on 2NRI3EB (Gamma-Metrics Wide Range neutron flux monitor channel B). Neutron monitor 2NRI3EB has two detectors. Each detector signal is connected to a separate Preamplifier (A3 and A4). This change will disconnect the A3 detector input to the A3 Preamplifier.

Reason for Activity:

The signal associated with this channel has been exhibiting spiking, and the source of this spiking has not been specifically identified and corrected. Operations and Reactor Engineering desire this channel to be available for refueling activities. Troubleshooting efforts have determined that the spiking may be attributed to a problem associated with the A3 detector, cabling, or connectors. Disconnecting the input to the A3 Preamplifier removes the noise input to the 'B' channel.

Effect of Activity:

This change affects the Gamma-Metrics (Post Accident) neutron flux monitors that are components of the Nuclear Instrumentation (NR) system. The effect of this change on the NR system will be to disconnect one fission chamber on Channel B of the Gamma-Metrics neutron flux monitors from its associated Preamplifier. The PANM system provides indication of neutron flux levels to the Main Control Room, the Plant Process Computer, and the Fire Hazards Panel. With the proposed change incorporated, changes in neutron flux will remain detectable. However, the absolute counts from the B channel will be lowered due to the absence of one detector. Changes in count rate are still detected. The A4 detector remains connected to the A4 Preamplifier, thus ensuring the 'B' channel of PANM continues to receive an input signal and thus maintaining channel functionality.

During operation at full power, no change in plant operation will occur as a result of this change. The wide range indication will continue to be provided from one (1) channel, and the narrow range indication will saturate, as it would even without the proposed change in place. During operation in Modes 2 though 6 and/or post accident conditions, Unit 2 Channel B narrow range will indicate approximately one half of the pulses seen on Unit 2 Channel A. Therefore, the absolute level will be impacted, but the relative level and the rate of change will continue to be valid.

Neither the design function of the neutron flux monitors or the NR system will be altered by this change.

Summary of Conclusion for the Activities 50.59 Review:

Since the outputs of the Preamplifiers (A3 and A4) are summed together to obtain the final output of the PANM monitors, and since the input to one Preamplifier will be disconnected, this change is considered adverse. Therefore, a 50.59 evaluation was performed.

NRC Reg. Guide 1.97 describes the requirements for the PANM channels. Specifically, PANM provide indications of changes in neutron flux during post-accident conditions.

In summary, the conclusion of the evaluation was that disconnecting one of the two neutron detectors of the Gamma-Metrics neutron flux monitor does not require NRC approval. Therefore, the proposed change may be implemented per applicable procedures.

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-2002-162**	**Rev.**	**0**
	50.59 Validation	**50.59 Validation No.**		**Rev.**	