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

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DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
SUMMARY OF FACILITY CHANGES, TESTS AND EXPERIMENTS AND SUMMARY OF COMMITMENT CHANGES

Pursuant to 10CFR 50.59(d)(2), enclosed is a summary description of Facility Changes, Tests and Experiments evaluated in accordance with 10 CFR 50.59(c) and implemented at the Kewaunee Power Station (KPS) during the last reporting period, which is defined as not to exceed 24 months.

A commitment change evaluation summary for those commitment changes that occurred during the last reporting period is also enclosed.

The enclosed summary encompasses all changes that occurred in both of the stated areas since our prior submittal dated June 3, 2010 (reference 1).

If you have questions or require additional information, please feel free to contact Ms. Mary Jo Haese at 920-388-8277.

Very truly yours,

Michael J. Wilson
Director Safety and Licensing
Kewaunee Power Station

Commitments made by this letter: NONE

Reference:

1. Letter from Michael J. Wilson (Dominion Energy Kewaunee, Inc.) to Document Control Desk (NRC), "Summary of Facility Changes, Tests and Experiments and Summary of Commitment Changes," dated June 3, 2010.

IE47
NRR

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ATTACHMENT 1

SUMMARY OF FACILITY CHANGES, TESTS AND EXPERIMENTS AND SUMMARY OF COMMITMENT CHANGES

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

10 CFR 50.59 Evaluations

50.59 Evaluation # 08-08-02

Activity Evaluated

DCR 3741 "Modify SI-350 A (B), add a Relief Valve between SI-350A(B) and SI-351A(B), and Change Gear Ratio to increase MOV valve disk DP limit"

Brief Description

DCR 3741 modified motor operated valves (MOVs) SI-350A (B) and added a relief valve between SI-350A (B) and SI-351A (B). These changes supported the resolution of a number of issues concerning the Safety Injection (SI) and Residual Heat Removal (RHR) systems including:

1. Air could have potentially been ingested into the RHR pumps during the initiation of containment sump recirculation.

2. The potential for Generic Letter (GL) 95-07 pressure locking of valves SI-350A and SI-350B.

3. The potential for GL 96-06 post LOCA and Main Steam Line Break (MSLB) inside containment pressurization of the pipe section between SI-350A and SI-351A and between SI-350B and SI-351B when the pipe section is water filled.

4. RHR system pressurization during a Small Break Loss of Coolant Accident (SBLOCA) could have potentially prevented containment sump recirculation valves SI-350A (B) from opening due to high differential pressure (DP) across the valves.

5. Pressurization of the water filled pipe between SI-350A and SI-351A and between SI-350B and SI-351B when heated by RHR/RCS contents on the downstream side of SI-351A (B) during RHR cool down alignment.

Reason for Change

The reason for DCR 3741 was to support the elimination of the air void in the pipe section between motor operated valves (MOVs) SI-350A and SI-351A and between SI-350B and SI-351B. The design change also addressed the potential increased differential pressure (DP) across SI-350A (B) caused by thermally induced pressurization associated with the RHR pumps operating with RCS pressure greater than the pump shutoff head under SBLOCA conditions.

The design change supported the resolution of a number of issues concerning the Safety Injection (SI) and Residual Heat Removal (RHR) systems.

Summary

A 10 CFR 50.59 evaluation was required due to the increased likelihood of malfunction of the SI-350A (B) MOVs. The likelihood of malfunction of the MOVs increased due to the allowed presence of water between SI-350 A (B) and SI-351A (B) and the associated reliance on the operation of relief valves SI-353A-2 and SI-353B-2 for the operation of SI-350 A (B).

The increase in the likelihood of malfunction for the MOVs was determined to be not greater than a factor of 2 and based on the guidance of revision 3 of the USA 10 CFR 50.59 Resource Manual the increase was considered to be minimal. Therefore, the modification did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the USAR.

10 CFR 50.59 Evaluation # 10-01-00

Activity Evaluated
DC KW-09-01010-000 – Reserve Supply Transformer (RST) and Spare RST Installation

Brief Description
The modification installed a Reserve Supply Transformer (RST) and its associated 138 kV and 20 kV equipment. This included installation of a spare RST (not normally connected to the offsite power system), RST load tap changer, circuit breakers, disconnect switches, associated protection and controls, a firewall to separate transformers, foundations, oil containment and catch basins, cabling and conduit, additional alarm and indication circuits, a new transmission monopole tower, and a new auxiliary station service transformer.

Reason for Change
The new RST supplies power to the new RAT, which was installed by DC KW-09-01011-000. These modifications improved reliability of offsite power to the station and improved operating and maintenance flexibility that can prevent entering a Technical Specification LCO action statement.

Summary
The 50.59 evaluation addressed the installation of new components (RST and new switchyard breaker RST199) that can increase the probability of a malfunction. The

evaluation concluded a no more than minimal increase in the likelihood of occurrence of a malfunction and a no more than minimal increase in the frequency of occurrence of an accident previously evaluated (loss of offsite power).

10 CFR 50.59 Evaluation # 10-02-01

Activity Evaluated

DC KW-09-01011-000 – Reserve Auxiliary Transformer (RAT) Replacement and Plant Interface

Brief Description

The modification installed a new Reserve Auxiliary Transformer (RAT) to replace the existing RAT. The new RAT is supplied by the RST installed by DC KW-09-01010-000. Additionally, the KPS substation-to-plant interface was modified. The interface for modification DC KW-09-01010-000, including the RST load tap changer (LTC) operated in Manual mode was installed. In addition, the Control Room interface for both the RST LTC and the existing TST LTC was installed. As part of this modification, the RAT fire protection deluge system was replaced and the RAT bay lighting was modified. Also, a new 30 foot manual sliding gate was installed to replace a 30 foot section of the nuisance fence on the Sally port at the Gatehouse to facilitate multiple transformer moves.

Reason for Change

The modification improved the reliability of offsite power to the station and improved operating and maintenance flexibility that can prevent entering a Technical Specification LCO action statement.

Summary

The 50.59 evaluation addressed the installation of a new component (RST LTC) that can increase the probability of a malfunction. The evaluation concluded a no more than minimal increase in the likelihood of occurrence of a malfunction and a no more than minimal increase in the frequency of occurrence of an accident previously evaluated (loss of offsite power).

50.59 Evaluation # 10-03-00

Activity Evaluated

DCR 3609-2 "Auxiliary Feedwater (AFW) Flow Control"

Brief Description

This modification added Motor Driven (MD) Auxiliary Feedwater (AFW) pump cross-connect piping, isolation valves, check valves and safeguards powered motor operated control valves with control room control and position indication to enhance operational flexibility. AFW pump discharge cavitating venturis were installed to provide passive protection of pumps from operation at runout conditions, thus allowing removal of all three AFW pump discharge pressure switches and associated pump trip and bypass circuits. New discharge check valves were added in the TDAFW pump piping to prevent flow to opposite train steam generators (SGs) from the MDAFW pumps. The AFW pump suction pressure trip actuation setpoints were revised to sequentially trip pumps and preclude tripping all pumps at one time thus preserving pump availability. MDAFW pump lube oil coolers were replaced and the piping modified to return condensate lube oil cooling water to the suction of the MD AFW pumps to conserve water. The Dedicated Shutdown Panel (DSP) was modified to reflect these changes.

The modification also added piping between the Safeguards Alley and the Turbine Building to permit AFW pump Service Water (SW) system pipe flushing operations without affecting the integrity of the associated barrier(s) during flushing.

The modification separated the CST supply piping to the Condenser from the CST supply piping to the AFW Pumps. The dedicated supply to the AFW Pumps reduces flow losses in the supply line to the AFW Pump.

Reason for Change

The purpose of this design change was to enhance the operation and reliability of the Auxiliary Feedwater (AFW) System. This was accomplished by providing passive pump protection features thereby reducing the number of active pump trip functions, and modifying pump protection features. These changes provide a more passively protected system that reduces the need to recover pump(s) that have been tripped off. The changes to the AFW system associated with this modification were elective; no feature of this modification was required to meet a regulatory commitment or address a design/licensing basis deficiency. This modification was the second and final phase of an overall AFW system improvement plan.

Summary

The Evaluation addressed the following changes that were determined to be adverse.

The added AFW MOVs are additional components that increase the likelihood of failure that could impact the AFW system design function to isolate AFW flow from a faulted/ruptured SG.

The added AFW MOVs' impact on the method of performing or controlling the design function to isolate a faulted/ruptured SG was determined to be adversely affected, because of the additional steps required to isolate the MDAFW pump cross-connect MOV(s)

The modification impacted the existing Kewaunee Steamline Break Mass/Energy Releases Outside Containment and Thermal Lag calculations such that the analyses were no longer bounding and required evaluation, therefore, this impact screened adverse.

The malfunction probability associated with the three components (one pump and two MOVs) was determined to be not more than two times the current malfunction probability associated with the two components (one pump and one MOV). Therefore the increase in the likelihood of malfunction was not more than minimal. The actions called out by emergency response procedures (secure a pump or close a valve) are not different than what the operators are already required to perform to isolate the affected steam generator, and are not new in technique. The combination of classroom, simulator and Operator time response validation provide the basis for concluding that there is no more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the USAR. The changes in harsh environment parameters remain within the limits of those parameters tested in the Qualification Test Reports for the impacted equipment, and the electrical equipment remains fully qualified to perform the required design functions. Therefore, there is no more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the USAR as a result of this change.

10 CFR 50.59 Evaluation # 10-04-00

Activity Evaluated
DC KW-09-01012-000 – American Transmission Company (ATC) Breakers Installation

Brief Description
The modification reconfigured the 345 kV substation as a double-bus, double breaker arrangement. In addition, a new 345/138 kV transformer T20 and 138 kV breaker was installed. The modification also included installation of new switchyard bus, surge arresters; coupling capacitor voltage transformers, foundations for new equipment; new protective relaying and control panels in the substation; connection to the new ground mat, a lightning mast and revised static wire lightning protection. Additionally, the generator output overhead line was rerouted from the G1 breaker to a new deadend structure in the 345 kV switchyard, and the existing switchyard DC system was modified to include two 125 VDC subsystems.

Reason for Change
The modification improved the reliability of offsite power to the station and improved

operating and maintenance flexibility that can prevent entering a Technical Specification LCO action statement.

Summary

The 50.59 evaluation addressed the installation of new components (345 kV switchyard breakers, 138 kV breaker, and 345/138 kV transformer T20) that can increase the probability of a malfunction. The evaluation concluded a no more than minimal increase in the likelihood of occurrence of a malfunction and a no more than minimal increase in the frequency of occurrence of an accident previously evaluated (loss of offsite power).

10 CFR 50.59 Evaluation # 10-06-00

Activity Evaluated

Design Change DCR 3669, Repower Control Room Air Conditioner (CRAC) Chiller Units

Brief Description

DCR 3669 repowered the non-safety related CRAC chiller units from safety related, safeguards diesel generator (SDG) backed power supplies to non-safety related, non-SDG backed power supplies. This design change also installed automatic transfer logic for the safety related cooling mode (known as "Alternate Cooling") to the Control Room from the Service Water system. Prior to DCR 3669 this mode of cooling was only available through manual actions.

Reason for Change

The purpose of this design change was to improve SDG loading margin and improve the available voltage on the two lowest margin safeguards motor control centers (MCCs). These MCC load and voltage improvements were needed during degraded grid voltage (DV) conditions just above the dropout setpoint of the DV relays coincident with a design basis accident. Implementing the changes aided in resolving the related operability determination and margin management issue on these components.

Summary

This 50.59 evaluation addressed the reduction in "defense-in-depth" of performing the Control Room cooling support function and revising the method of controlling this cooling support function. This evaluation also addressed the additional step necessary to align "Alternate Cooling" following an Appendix R fire event.

The evaluation concluded a no more than minimal increase in the likelihood of occurrence of a malfunction. Although SDG powering of the Control Room CRAC chiller units has been made unavailable by repowering the non-safety related chillers from non-safety related, non-SDG backed power supplies, the reliability of the new automatic transfer logic for "Alternate Cooling" is maintained through the use of redundant (properly separated and meeting single failure criteria), qualified components and periodic testing and calibration. Operator ability and response time during an Appendix R fire event were not impacted by the additional step to manually align "Alternate Cooling" to the Control Room.

10 CFR 50.59 Evaluation # 10-07-00

Activity Evaluated

Engineering Technical Evaluation (ETE) ETE-NAF-2010-0075 "Implementation of Revised Safety Analysis Limit for Low Steam Line Pressure SI Setpoint for KPS"

Brief Description

The activity involved implementation of a revised safety analysis limit (SAL) for the low steam line pressure SI trip setpoint. This SI trip is credited in the analysis of several events including Excessive Heat Removal due to Feedwater System Malfunction, Main Steamline Break and High Energy Line Break. The analysis bases for these events were reviewed in the ETE.

Specific cases of the Excessive Heat Removal due to Feedwater System Malfunction event were reanalyzed with the revised setpoint. The remaining USAR safety analyses that credit low steam pressure SI were evaluated in ETE-NAF-2010-0075 and found to be unaffected. The USAR Chapter 14 safety analyses were shown to satisfy the applicable acceptance criteria with the revised SAL.

The decrease in low steam line pressure SI SAL created additional margin from the Technical Specifications engineered safety feature setting limit and supported the use of Methods 1 or 2 as detailed in Part II of ISA Standard S67.04 for applying the CSA. The change also supported the implementation of KPS Improved Technical Specifications (ITS).

Reason for Change

Revising the SAL for the low steam line pressure SI setpoint provided margin to eliminate a non-conservatism between the As Found Tolerance and the Allowable Value for the SI setpoint which had been noted in Technical Report EE-0116 Revision 7. The SAL revision and associated ETE eliminated the need to perform a plant setpoint change prior to implementing Improved Technical Specifications.

Summary

The 50.59 evaluation addressed implementation of a reduction in the low steam line pressure SI SAL. This change was considered to have an adverse impact on the design functions of the safety injection system to 1) deliver borated water to the RCS to add shutdown reactivity, 2) initiate Feedwater isolation and 3) initiate steam line isolation.

Implementation of a new Excessive Heat Removal due to Feedwater System Malfunction analysis case is not an initiator of an accident and no new failure modes were introduced. No physical changes were made to KPS systems, structures, or components (SSC). The new analysis case relied upon the same SSCs as the existing analysis case and used the same evaluation method. It was determined that this activity did not result in more than a minimal increase in the consequences of an accident previously evaluated. No new malfunctions or failure modes were created by this change.

50.59 Evaluation # 10-08-00

Activity Evaluated

Updated Safety Analysis (USAR) Change Request KPS-UCR-2010-061, Revise Auxiliary Feedwater (AFW) Pump Capability Statements

Brief Description

50.59 Evaluation 10-08-00 was prepared in support of a revision to the USAR to revise the AFW pump capability statements to indicate the turbine driven AFW (TDAFW) pump reaches full flow within 90 seconds. Prior to making this change, the USAR indicated all AFW pumps, motor driven and turbine driven, reach full flow within one minute. The full flow capability for the motor driven AFW pumps was not changed and remains as 60 seconds (one minute).

Reason for the Change

Surveillance test results indicate the TDAFW pump often does not reach full flow within one minute, but is capable of reaching full flow within 90 seconds. Ninety seconds was chosen to allow margin for statistical variation in the test results and more accurately reflect pump performance.

Summary

The 50.59 Evaluation addressed allowing a longer time for the TDAFW pump to reach full flow. This change was determined to be adverse for events requiring timely AFW flow delivery by the TDAFW pump. The limiting Loss of Normal Feedwater (LONF) event (LONF without loss of AC power) and Station Blackout (SBO) event were determined to require timely delivery of AFW using the TDAFW pump.

It was determined that if the TDAFW pump is credited for the LONF event, such as due to a single failure of a motor driven AFW pump, the event can be mitigated with the TDAFW pump delivering flow within 90 seconds. The flow delivery by use of one motor driven AFW pump (within 60 seconds) along with the TDAFW pump (within 90 seconds) is bounded by the analysis using two motor driven AFW pumps (within 60 seconds) due to ample AFW flow margin in the analysis.

The TDAFW pump is credited for four hour SBO coping. It was determined that allowing 90 seconds for the TDAFW pump to reach full flow will not prevent the ability of the TDAFW pump to deliver the required 41,500 gallons of AFW to the steam generators during the four hour SBO coping period.

It was determined that this change did not result in more than a minimal increase in the consequences of an accident previously evaluated. No new malfunctions, malfunction results, or new accidents were created by this change.

10 CFR 50.59 Evaluation # 11-01-00

Activity Evaluated

Design Change KW-10-01177, Bus 1-3 and Bus 1-4 Fast Bus Transfer Changes

Brief Description

KW-10-01177 modified the main auxiliary transformer (MAT) to reserve auxiliary transformer (RAT) bus transfer scheme for 4160V buses 1-3 and 1-4 by removing a permissive in the RAT supply breaker CLOSE circuit that indicates the MAT supply breaker is open. The effect of this change is a faster transfer from the MAT to the RAT and decreased bus dead time since the RAT breaker will be closing while the MAT breaker is opening (parallel actions) instead of these actions occurring in series.

Reason for Change

Analysis determined that the MAT-to-RAT transfer on 4160V buses 1-3 and 1-4 does not meet its acceptance criteria. The physical implications of not meeting the acceptance criteria included possible equipment damage, particularly to pump/motor shafts, due to excessive forces generated during out-of-phase power transfers. These physical implications were further exacerbated by another design change that was installing variable frequency drives on the Heater Drain pumps.

Summary

This 50.59 evaluation addressed the increased probability of losing an offsite power supply (via the RAT) with the proposed configuration. The evaluated event involved losing the RAT power supply due to failure of one of the MAT supply breakers to buses 1-3 or 1-4 to OPEN during an event requiring a fast bus transfer from the MAT to the RAT under the proposed configuration. The function of the RAT is to provide power to downstream systems and components to support mitigation of design basis accidents and events, if the RAT remains available.

The evaluation concluded that prior NRC approval was not required. The RAT (and offsite power) is not an initiator of any accidents. The RAT power supply supports downstream systems and components during accidents and events only if it is available. The likelihood of occurrence of a malfunction has not been more than minimally increased. It was determined that several factors beyond failure of the additional two MAT supply breakers to buses 1-3 and 1-4 maintain the likelihood of occurrence of a malfunction less than a factor of two and therefore not more than minimal. The proposed configuration and changes to the breakers do not increase their probability of failure.

10 CFR 50.59 Evaluation # 11-03-00

Activity Evaluated

Engineering Technical Evaluation (ETE) ETE-NAF-2011-0020 "Implementation of Changes to the Kewaunee Main Steam Line Break (MSLB) Analysis Basis for Reduced Safety Injection Flow and Offsetting Effects"

Brief Description

The activity involved implementation of changes to the Kewaunee Main Steam Line Break (MSLB) accident analysis. The new MSLB analysis (1) implemented a new, conservative safety injection flow curve, (2) credited a Refueling Water Storage Tank (RWST) boron concentration consistent with the Kewaunee Power Station (KPS) Technical Specifications, (3) incorporated a dynamic Internal Containment Spray (ICS) model which accounts for the difference between containment pressure and RWST liquid head, and (4) credited a change to the ICS header nozzle plugging requirements which increases the number of nozzles available to provide ICS flow to containment.

Reason for Change

A revised MSLB accident analysis was needed to address a new safety injection flow curve. The new safety injection flow curve was conservative compared to the curve used in the existing USAR analyses.

Summary

The 50.59 evaluation addressed implementation of the new safety injection flow curve in the MSLB analysis. This change was considered to have an adverse impact on the design function of the safety injection system to deliver borated water to the RCS to add shutdown reactivity to the core in the event of a MSLB accident because the new safety injection flow curve was conservative compared to the analysis curve used in the existing USAR analyses.

Implementation of a new MSLB containment response analysis case is not an initiator of an accident and no new failure modes were introduced. No physical changes were made to KPS systems, structures, or components (SSC). The final analytical values required for the SSCs involved in the analysis were within the SSC design limits and were conservative with respect to current plant test data. The new analysis case relied upon the same SSCs as the existing analysis case and used the same evaluation method. It was determined that this activity did not result in more than a minimal increase in the consequences of an accident previously evaluated. No new malfunctions or failure modes were created by this change.

Commitment Change Evaluation Summary

Document(s) Evaluated:

1. Letter from P. D. Ziemer (WPSC) to USNRC dated May 11, 1983, title: "Response to Notice of Violation and Proposed Imposition of Civil Penalty"
2. Letter from D.C. Hintz (WPSC) to USNRC dated June 3, 1988, title: "Generic Letter 88-05: Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"

Brief Description:

In the first letter, KPS committed to periodic inspections of containment by the Plant Manager or one of his direct reports. This was in response to an event in 1982 wherein all Containment Pressure instrument sensing lines inside containment were found capped, thereby rendering them inoperable. It was stated that this action would "...keep the plant management informed of the condition of the plant."

In the second letter, KPS responded to a request for confirmation that a program was in place to ensure that boric acid corrosion would not lead to degradation of the RCS pressure boundary. KPS cited management inspections of containment as a component of this program.

Scope: Personnel designated to perform containment inspections.

Basis for Change:
Regarding the first letter: It was determined that this commitment would be an ineffective barrier against events similar to the initiating event. It would only detect such a condition after occurrence. The other actions taken in response (labeling, checklist and procedure revisions) and modern standards for operation and configuration control have been effective in preventing recurrence. The standard expectation for reporting of discrepancies (Condition Reporting) and the routing requirements of the inspection report serve to keep plant management informed of the condition of the plant.

Regarding the second letter: The KPS response did state that the containment inspections are performed by management. However, because no reason is given to explain why management, instead of other personnel, performs the inspection, it is likely that this was a statement of existing fact, not a requirement. Operations personnel are more suited to the task of leakage identification and quantification.

Summary: The commitment: "The Plant Manager—Nuclear or direct reports shall perform the Quarterly Containment Inspection when the plant is at full power" has been revised to "A qualified Auxiliary Operator, Reactor Operator, or Senior Reactor Operator shall perform the Quarterly Containment Inspection when the plant is at full power."