10 CFR 50.59(d)(2)

Progress Energy

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Ladies and Gentlemen:

Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc. (PEC), submits the attached report in accordance with 10 CFR 50.59(d)(2), "Changes, Tests, and Experiments," for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The report provides a description of changes that were implemented pursuant to 10 CFR 50.59 between April 1, 2006, and April 1, 2008. A summary of the evaluation for each item is also included in the attached report.

If you have any questions concerning this matter, please contact me at (843) 857-1626.

Sincerely,

Curt Castell

Curt Castell
Supervisor – Licensing/Regulatory Programs

CAC/cac

Attachment

c: V. M. McCree, NRC, Region II
NRC Resident Inspector, HBRSEP
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SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS FOR THE H. B. ROBINSON STEAM ELECTRIC PLANT (HBRSEP), UNIT NO. 2

Evaluations performed in accordance with 10 CFR 50.59 for the time period of April 1, 2006, to April 1, 2008:

Evaluation No. 194735:

Description:

This evaluation pertains to the engineering change to the facility to install a selector switch in the circuit between the general alarm relay for the "D" instrument air compressor (IAC) and the annunciator to allow operations to break the circuit, thereby manually acknowledging the "D" IAC status, and disabling the Window E-7 alarm in the control room.

Summary of Evaluation:

Section 9.3.1.3 of the Updated Final Safety Analysis Report (UFSAR) states that the instrument air system has no functions to perform for safe shutdown or accident conditions. It is therefore classified as a non-safety system. The only parts of the system that are Class I are the containment penetration and the associated isolation valves and auxiliary air accumulators with connected tubing/piping for the main steam isolation valve operators. The modification to install a selector switch does not affect the required containment isolation function. This modification does not increase the frequency of occurrence of an accident, increase the likelihood of occurrence of a malfunction of a system, structure, or component (SSC) important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR, because the instrument air system operation will not be affected by the change to the alarm circuit.

Additionally, this modification does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. This change does not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered, because the consequences of this change remain bounded by existing accident analyses. This change does not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses. The configuration created by this change remains fully bounded by existing analyses using current evaluation and safety analysis methodologies.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 198887:

Description:

This evaluation pertains to the changing of the environmental qualification (EQ) profiles for pressure and temperature due to a change in peak pressure and temperature results for the Loss of Coolant Accident (LOCA) Containment analysis. A bounding set of profiles has been established. Also, the set-point for the Isolation Valve Seal Water (IVSW) low pressure alarm was changed. The new limiting case for the LOCA Containment analysis and the Technical Specification amendment to change the containment peak accident pressure (Pa) were submitted to the NRC and approved by Amendment No. 215 to the operating license. The change to the IVSW alarm set-point will preserve the design function of the IVSW system, and will not impact its performance.

Summary of Evaluation:

The new LOCA Containment analysis produced different results based on correcting various deficiencies in the modeling scheme of implementation of input parameters. Nothing has been altered that would change the frequency or contribute to the initiation of the LOCA event. Altering the set-point for minimum IVSW pressure requirement will re-establish the design function and the intent of the IVSW system post initiation of a LOCA event. The new EQ profiles only impact the EQ evaluations. Therefore the changes being performed do not impact any initiating circumstances as they relate to a LOCA event and do not impact the frequency of occurrence.

The change to the set-point for minimum IVSW pressure requirement is within the range of operation of IVSW system, which is normally operated above the newly established set-point. Sufficient margin exists between the set-point, the minimum required pressure for IVSW and the normal system operation. The new EQ profiles are adequate for the equipment that falls under the EQ program and do not exceed any of the equipments most severe design basis accident condition requirements.

The new LOCA Containment analysis pressure is bounded by the newly established Pa value and the leak rates measured to be less than the dose consequence analysis design input leak rates, therefore the consequences are not altered. The equipment important to safety will perform as intended to preserve their design function. The IVSW system will perform as intended and EQ equipment will have adequate longevity in a post LOCA environment. The changes will not impact any of the initiating causal factors associated with possible equipment malfunction or failure. The EQ profiles and the IVSW minimum alarm set-point are not directly related to the analyses of any fission product barrier.

The new analysis does not deviate from the methodologies previously described. The manner in which the design inputs were modeled in the previous analysis have changed and therefore, the new analysis for LOCA Containment was submitted to the NRC and approved by Amendment No. 215 to the HBRSEP, Unit No. 2, operating license. The changes associated with the EQ profiles and IVSW are in direct support of the implementation of the analyses changes.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 214645:

Description:

This evaluation pertains to a revision to the end-of-life (EOL) moderator temperature coefficient (MTC) operating (from about -40 pcm /°F to about -45 pcm/°F) and surveillance limits. These limits are provided in the Core Operating Limits Report (COLR). This evaluation also includes the analysis of increased volume of unborated water in the safety injection (SI) lines.

Summary of Evaluation:

The analyses performed support operation with the revised EOL MTC and with increased unborated volume in the SI lines. The analyses included a disposition of Chapter 15 events; the events that required evaluation are as follows:

- Increase in Steam Flow (UFSAR 15.1.3)
- Main Steam Line Break (UFSAR 15.1.5)
- Uncontrolled RCCA Bank Withdrawal at Power (UFSAR 15.4.2)
- Withdrawal of a Single Full-Length RCCA (UFSAR 15.4.3.1)
- Dropped RCCA/RCCA Bank (UFSAR 15.4.3.3)

The analyses demonstrate that the affected events continue to meet their acceptance criteria; the consequences are unaffected by the proposed changes. There are no new accidents or new SSC failure mechanisms introduced by the proposed activities.

The analyses performed in support of the EOL MTC operating limit (-45 pcm/°F) were performed using the S-RELAP5 computer code. The use of the S-RELAP5 methodology has been reviewed and approved by the NRC for HBRSEP, Unit No. 2. The S-RELAP5 non-LOCA methodology has been added to the Core Operating Limits Report to be consistent with the revised Technical Specifications, in accordance with License Amendment No. 211, as approved by the NRC by letter dated November 29, 2006. The use of the new methodology is not a departure from a method of evaluation as described in the UFSAR because the new methodology has been reviewed and approved by the NRC.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 214818:

Description:

This evaluation pertains to a revision to the Offsite Dose Calculation Manual as summarized in the following information.

Revised Table 2.8-1 (Radioactive Liquid Waste-Sampling and Analysis Program) including changed the frequency for sampling continuous releases for composite purposes from "D" (once per 24 hours) to 3 times per week, changed the frequency for performing I-131 analysis on continuous releases from once per month to three (3) times per week, added note "h" to state that tritium analyses on batch tank releases may be performed on an individual basis instead of performing an analyses on a composite sample, and added note "i" to state that continuous releases samples may be analyzed on individual samples verses composite samples, if desired.

Revised Table 3.10-1 (Radioactive Gaseous Effluent Monitoring Instrumentation) Section 1.e (Plant Vent Sample Flow Rate Monitor) to change the minimum channels operable from "1 of the 2 monitors" to "1."

Revised Table 4.1-2 (Reporting Levels For Radioactivity Concentrations In Environmental Samples) to change the reporting level for I-131 in water from 2 pCi/l to 20 pCi/l.

Revised Table 4.1-3 (Lower Limits of Detection [LLD]) to delete the LLD value for gross beta analysis in water.

Added new Section 8.3, which describes requirements associated with compensatory actions.

Revised Section 9.1.4, which details the requirements for solid waste reporting in the Radioactive Effluent Release Report.

Summary of Evaluation:

These ODCM changes did not increase the frequency of occurrence of an accident, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, these ODCM changes did not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. These changes do not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered. These changes do not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 222649:

Description:

This evaluation pertains to analyses and procedure changes to allow removal of the containment equipment hatch earlier than the current limit of 67 hours after shutdown. The changes were evaluated to ensure that commitments related to mitigating strategies for a loss of decay heat (shutdown cooling) event are not compromised.

Summary of Evaluation:

Due to the redundancy in the Residual Heat Removal (RHR) system, the Component Cooling Water (CCW) system, and the Service Water (SW) system, a complete loss of shutdown cooling is outside of the HBRSEP, Unit No. 2, design and licensing basis and is not analyzed in the UFSAR. However, commitments have been made to comply with the mitigating strategies in NUMARC 91-06 in response to an assumed loss of shutdown cooling event. One of the commitments is a requirement to ensure that the containment, including the equipment hatch, is closed prior to the onset of boiling within the core. The time that this can be accomplished is being reduced from 67 hours to 24 hours after shutdown. Removing the equipment hatch at 24 hours will allow for shorter refueling outages.

One of the procedural changes incorporated is the timelier initiation of Safety Injection (SI) following a loss of shutdown cooling. Calculations and time validations of procedural changes demonstrated that SI initiation can delay boiling in the core until after the equipment hatch can be reinstalled. To support the timelier initiation of SI, there was also a change in the manner in which the SI pump is made incapable of injecting while at low temperature conditions.

These changes in the use of SI and the manner in which it is isolated at low temperature were evaluated with regard to the requirements for low temperature over pressure (LTOP) protection of the reactor vessel to prevent brittle fracture. It was concluded that an LTOP vessel fracture event was not credible. Since there is no increased potential for vessel fracture, there is no increase in the frequency or likelihood of any accident or malfunction of equipment important to safety currently evaluated in the UFSAR, and no new accidents or failures not previously evaluated are created. The revised procedural controls will continue to ensure that boiling in the core will not occur until after containment closure is achieved. Therefore, there will be no increase in radiological consequences and the design basis limits for the cladding, RCS, and containment will not be exceeded. Since a complete loss of all shutdown cooling is outside of the HBRSEP, Unit No. 2, design bases and is not considered in the safety analyses, the method of evaluating this event does not result in a departure from an evaluation methodology used to establish the design basis or used in the safety analyses.

As described, these changes did not increase the frequency of occurrence of an accident, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, these changes did not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. These changes do not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered. These changes do not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

<u>Evaluation No. 226110</u>:

Description:

This evaluation pertains to procedure changes to LP-551, "Rod Position Indication System," that will require a reactor coolant system (RCS) boron concentration that will ensure that the core is approximately 2000 pcm subcritical (Keff = 0.98) with all control rods withdrawn for the rod position indication (RPI) calibration. Also, the previous practice of withdrawing only one control rod bank at a time for RPI calibrations involved the selection of the individual bank on the Rod Selector Switch has been changed to allow the Control Banks to be withdrawn in "Manual" which allows the movement of more than one bank of control rods at a time.

Summary of Evaluation:

The boron concentration required by LP-551 ensures that both the Shutdown Margin requirements of Technical Specification 3.1.1 and the Core Operating Limits Report (COLR) and the subcriticality requirements of Mode 3 (Keff < 0.99) continue to be met. The amount of Shutdown Margin and core subcriticality are used as inputs to the evaluations of accidents and malfunctions of SSCs important to safety but do not act as the initiators for any accidents or malfunctions of equipment important to safety evaluated in the UFSAR. The reduction in the amount of core subcriticality does not alter the interface of the reactor core with any nuclear instrumentation system (NIS), reactor protection system (RPS) or engineered safety feature actuation system (ESFAS) function. Since the NIS, RPS and ESFAS design functions are not affected and the Shutdown Margin requirements are met, the reduction in the amount of subcriticality during the RPI calibration does not increase the consequences of any accident or malfunction of SSCs important to safety evaluated in the UFSAR. Accidents and malfunctions that could occur due to the use of "Manual" to move the Control Banks in overlap have already be evaluated in the UFSAR and no conditions established during the RPI calibration will increase frequency of occurrence of these analyzed accidents and malfunctions. No new accidents or malfunctions of SSCs important to safety are created due to the fact that no plant conditions or configurations will be created that are significantly different than normal operation in Modes 1 and 2. Since there is no adverse impact on the RPS and ESFAS functions and Shutdown Margin is maintained, there will be no adverse change in the integrity of the fission product barriers. The method used to ensure adequate Shutdown Margin is consistent with the approach and considerations described in the bases of Technical Specification 3.1.1 and uses data that is generated by the fuel vendor using the NRC approved methods identified in Technical Specification Section 5.6.5.

As described, these changes did not increase the frequency of occurrence of an accident, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, these changes did not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. These changes do not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered. These changes do not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

<u>Evaluation No. 228385</u>:

Description:

This evaluation pertains to a temporary modification to the Fire Detection and Actuation Panel (FDAP) A1, FDAP A2, and the Train A fire detection transceivers. UFSAR Section 9.5.1B.E.1.a states: "The low-voltage fire detection systems are connected to emergency power buses, and is provided with four-hour battery backup." The temporary 120Vac power is from a normal plant power bus, not an emergency power bus.

Summary of Evaluation:

This temporary modification provided 120 Vac power from normal plant power buses to the FDAPs and transceivers while the Train-A emergency power bus was deenergized for maintenance. The temporary power was supplied from lighting panels to the fire detection loads via standard electrical receptacles and a plug-in power cable. The temporary power was expected to be needed and connected for approximately 12 hours. Should a loss of power occur to the lighting panel or receptacle, the temporary power cable could have been plugged into another plant receptacle if it was energized. So the reliability of the 120 Vac was only minimally less reliable than the emergency power supply. The FDAPs and transceivers have four-hour battery back-ups which would supply power until the 120 Vac power is restored or give time to establish the required fire watches as compensatory measures per FP-012. These actions on a loss of 120 Vac are the same whether the 120 Vac power source is emergency bus supplied or from a normal power bus. Also the Train-B fire detection system will still be powered from an emergency power bus maintaining its normal full capability. The likelihood of a malfunction to the Train-A Fire Detection and Actuation System due to the loss of temporary power is insignificantly more likely than a malfunction due to a loss of emergency bus power during the short time that temporary power is connected to the Train-A Fire Detection and Actuation System.

The FDAPs and transceivers are designed to work on 120 Vac power. The temporary modification supplied 120Vac, and the temporary power is coming from lighting panels via standard electrical receptacles that are designed to provide power to loads on a temporary basis; therefore, there was no increase in the likelihood of a malfunction to the electrical distribution system due to temporary power being connected to the FDAPs and transceivers.

As described, these changes did not increase the frequency of occurrence of an accident, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, these changes did not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. These changes did not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered. These changes do not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses.

This change does not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses. The plant design and associated design description remain fully bounded by existing analyses using current evaluation and safety analysis methodologies.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 230553:

Description:

This evaluation pertains to changes to the Systems Operations Department procedures that provide the voltage schedule for the plant off-site power systems (i.e., switchyards). Section 8.2.2 of the UFSAR specifies the per unit values of the switchyard voltage schedule. The existing voltage schedule is based on a target value and the revised voltage schedule utilizes a voltage range with a target value.

Summary of Evaluation:

Loss of off-site power (LOOP) is a bounding condition for the proposed procedure changes. The proposed range in the voltage schedule is not an accident initiator. The electrical distribution system is utilized to support the operation of plant equipment essential to the prevention, or to the mitigation of the consequences, of nuclear accidents. The change in voltage schedule does not cause the electrical distribution system or the off-site power sources to operate outside their current design or testing limits. The existing voltage schedule is based on a single target value and the revised voltage schedule utilizes a voltage range with a target value. Applicable calculations continue to demonstrate that the Off-site Power System distribution equipment has been adequately sized to meet the voltage requirements imposed by the system and its electrical loads. Electrical protective features (i.e., degraded grid voltage relay) of safety related electrical equipment and their operation are not impacted by the subject change. Therefore, the proposed activity does not result in an increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

Design, material, and construction of existing plant components are not changed by this activity. The changes in the voltage schedule do not cause the electrical distribution system or the plant switchyards to operate outside their current design or testing limits. The changes do not prevent actions credited for accidents involving electrical loading considerations. Operator action/guidance is governed by NGGM-IA-0003, "Transmission Interface Agreement for Operation, Maintenance, and Engineering Activities at Nuclear Plants," and the subject System Operations Department procedures. The existing switchyard voltage schedule is based on a single target value and the revised voltage schedule utilizes a voltage range with a target value. The proposed voltage schedule maintains switchyard voltages above the Systems Operations minimum required Robinson switchyard voltage to ensure proper operation of emergency loads following a Loss of Coolant Accident (LOCA) or a unit trip. Plant operator response during design basis accident mitigation does not change as a result of the proposed voltage schedule. Therefore, the proposed activity does not result in an increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

As described, these changes do not increase the frequency of occurrence of an accident, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, these changes do not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. These changes do not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered. These changes do not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 233428:

Description:

The Systems Operations Department procedure SORMC-GD-17 provides the voltage schedule for the plant off-site power systems (i.e., switchyards). Section 8.2.2 of the UFSAR specifies the per unit values of the switchyard voltage schedule. The voltage schedule in Section 8.2.2 of the UFSAR is based on a target value only and the revised voltage schedule utilizes a voltage range with a target value.

Summary of Evaluation:

Loss of off-site power (LOOP) is a bounding condition for the proposed procedure changes. The proposed range in the voltage schedule is not an accident initiator. The electrical distribution system is utilized to support the operation of plant equipment essential to the prevention, or to the mitigation of the consequences, of nuclear accidents. The change in voltage schedule does not cause the electrical distribution system or the off-site power sources to operate outside their current design or testing limits. The existing voltage schedule is based on a single target value and the revised voltage schedule utilizes a voltage range with a target value. Applicable calculations continue to demonstrate that the Off-site Power System distribution equipment has been adequately sized to meet the voltage requirements imposed by the system and its electrical loads. Electrical protective features (i.e., degraded grid voltage relay) of safety related electrical equipment and their operation are not impacted by the subject change. Therefore, the proposed activity does not result in an increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

Design, material, and construction of existing plant components is not changed by this activity. The changes in the voltage schedule do not cause the electrical distribution system or the plant switchyards to operate outside their current design or testing limits. The changes do not prevent actions credited for accidents involving electrical loading considerations. Operator action/guidance is governed by NGGM-IA-0003, "Transmission Interface Agreement for Operation, Maintenance, and Engineering Activities at Nuclear Plants," and the subject System Operations Department procedures. The existing switchyard voltage schedule is based on a single target value and the revised voltage schedule utilizes a voltage range with a target value. The proposed voltage schedule maintains switchyard voltages above the Systems Operations minimum required Robinson switchyard voltage to ensure proper operation of emergency loads following a Loss of Coolant Accident (LOCA) or a unit trip. Plant operator response during design basis accident mitigation does not change as a result of the proposed voltage schedule. Therefore, the proposed activity does not result in an increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

As described, these changes do not increase the frequency of occurrence of an accident, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, these changes do not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. These changes do not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered. These changes do not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 236180:

Description:

This evaluation pertains to the core loading pattern and associated analyses in support of the Cycle 25 operation including new reload batch neutronic design, new core loading pattern, safety analysis, and updates to the UFSAR and Plant Operating Manual (POM) procedures.

Summary of Evaluation:

Operation with a reload batch of 57 fresh natural uranium axial blanket (NUAB) fuel assemblies was evaluated. The ROB2-25 (Region 28) NUAB fuel assemblies contain gadolinia-bearing fuel rods. All 57 assemblies contain High Thermal Performance (HTP) spacers, Intermediate Flow Mixer (IFM) grids, and FUELGUARD debris-resistant lower tie plates. Cycle 25 is the sixteenth core reload with NUAB assemblies and the sixteenth successive reload containing gadolinia-bearing fuel.

The safety analyses support Cycle 25 operation at a nominal core power level of 2339 MWt for up to 516 effective full-power days (EFPD). The Cycle 25 safety analyses are based on a Cycle 24 shutdown between 497 EFPD and 527 EFPD. The following areas were evaluated to support Cycle 25: mechanical evaluation, neutronics evaluation, thermal hydraulic evaluation, setpoints verification, Chapter 15 safety analyses, and several non-Chapter 15 safety analyses.

The fuel mechanical design was evaluated and the results support operation to a maximum assembly exposure of 57.0 Gigawatt-day per metric ton uranium (GWd/MTU) and a maximum rod exposure of 62.0 GWd/MTU. The characteristics of the fuel and the reload core were verified to be in conformance with the current Technical Specification limits.
The thermal hydraulic compatibility of all the core assemblies is ensured because the Cycle 25 core will consist of only HTP/IFM fuel. No mixed core penalty will be applied to the Minimum Departure from Nucleate Boiling Ratio (MDNBR) limit for Cycle 25.

The potential for rod bow effects was evaluated and it was determined that no rod bow penalty on MDNBR or peak Linear Heat Generation Rate (LHGR) is required. The effect of DNB propagation on fuel failure is reflected in the results for Cycle 25. The analysis of record Alternative Source Term (AST) LOCA doses are bounding for Cycle 25. For the radiological consequences of the non-LOCA events, it was verified that the parameters applied in the AST analysis are bounding of the same parameters for Cycle 25. The Chapter 15 and several non-Chapter 15 safety analyses were reviewed with respect to Cycle 25 plant configuration/operation and neutronics changes. The event review indicated that due to changes in neutronic characteristics some events required a complete or partial (e.g., MDNBR or fuel centerline melt) re-analysis for Cycle 25. Other events were re-performed to correct inputs errors discovered since the completion of the Cycle 24 reload. The re-analysis and evaluation confirmed that all applicable acceptance criteria continue to be met for each event.

The Misload event (UFSAR15.4.7) analysis concluded that misload scenarios that can lead to fuel failure are detected either during the initial 30% flux map or during routine Technical Specification surveillances of core power distribution. Misload scenarios involving the interchange of GG57 (a fresh assembly for Cycle 25) and a once burned fuel assembly cannot be detected in the initial 30% flux map. The safety analysis demonstrates that routine Technical Specification surveillance (3.2.1.1 and 3.2.2.1) can detect the subject GG57 misload scenarios prior to causing fuel failure.

The overtemperature delta-T and overpower delta-T trip functions were re-analyzed. The results verified that the trip functions and setpoints provide sufficient protection for Cycle 25.
As described in the above discussion of the acceptability of the analysis results, implementation of the Cycle 25 core design and supporting safety analyses demonstrated that the requirements

and acceptance criteria defined in the UFSAR are satisfied for Cycle 25 operation. Therefore, the Cycle 25 reload design, with regard to the safety analysis, will continue to meet the plant licensing basis.

As described, these changes did not increase the frequency of occurrence of an accident, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, these changes did not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. These changes do not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered. These changes do not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

<u>Evaluation No. 240975</u>:

Description:

This evaluation pertains to a UFSAR change in which Section 8.3.3 of the UFSAR specifies that cable loading criteria were developed using IPCEA Publication P-46-426 is revised to specify IPCEA P-54-440. The revise reference is more appropriate for assessing cable adequacy in cable trays.

Summary of Evaluation:

The proposed activity incorporates a previous commitment by Carolina Power & Light (CP&L) to the NRC. The revised reference provides additional assurance that electrical cables are capable of performing their design function. Loss of off-site power is a bounding condition for the proposed activity. The proposed activity incorporates additional criteria for sizing cables. The electrical distribution system is utilized to support the operation of plant equipment essential to the prevention, or to the mitigation of the consequences, of nuclear accidents. The change in loading criteria does not cause the electrical distribution system or the off-site power sources to operate outside their current design or testing limits. Electrical protective features of safety related electrical equipment and their operation are not impacted by the subject change. Therefore, the proposed activity does not result in an increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

Design, material, and construction of existing plant components are not changed by this activity. Evaluations for cable trays in Calculation RNP-E-5.001 require the most conservative sizing of IPCEA P-46-426 and IPCEA P-54-440. The proposed activity provides additional current sizing criteria that are equal to or better than the existing sizing criteria. The additional criteria in cable sizing do not cause the electrical distribution system equipment to operate outside their current design or testing limits. Therefore, the proposed activity does not result in an increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

As described, these changes did not increase the frequency of occurrence of an accident, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, these changes did not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. These changes do not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered. These changes do not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

<u>Evaluation No. 244567</u>:

Description:

This evaluation pertains to a plant modification that permanently defeats power mismatch and RCS pressure compensation inputs to the automatic rod control system. This change affects the UFSAR-described design of the system in that the system will be less responsive and design features described in the UFSAR will no longer be required, nor available.

Summary of Evaluation:

The changes to the automatic rod control system are designed to reduce the amount of spurious rod steps due to normal plant system fluctuations. Rod control is identified as the initiating system for events in Chapter 15 (15.4.1, 15.4.2, 15.4.3), the changes being made eliminate potential causes of rod movement without creating new failure modes or effects. Therefore, there is no increase in the frequency of occurrence of an accident.

The changes reduce the number of inputs to the automatic rod control system, thereby decreasing the chances of a malfunction of the system. As mentioned above, no new failure modes or effects are being created; in fact, one failure mode leading to unwarranted automatic rod motion (N-44 failure) is being eliminated. NRC Generic Letter 89-19 addresses concerns with control system failures resulting in plant transients that could challenge the reactor protection system or other protective components. The proposed changes do not invalidate the conclusions of Generic Letter 89-19 or the HBRSEP, Unit No. 2, response to the Generic Letter. NUREG-0737, Item II.K.3.2, required licensees to determine the probability of a Small-Break Loss of Coolant Accident due to a stuck open Pressurizer Power Operated Relief Valve caused by plant transients. These changes do not invalidate the HBRSEP, Unit No. 2, response to NUREG-0 737, Item II-K.3.2, as documented in WCAP-9804. Therefore, there is no increase in the likelihood of a malfunction of an SSC important to safety.

The automatic rod control system is provided to the operator for improved response during plant transients. As noted in UFSAR 7.7.1, the automatic control systems such as automatic rod control, pressurizer level and pressure control, steam dump and Feedwater control are designed to limit plant parameters for the designed load perturbations (10% step change and 5% per minute ramp change) so that reactor trips will not occur for these load changes. The evaluations in Engineering Change (EC) 67518 demonstrate that the automatic control systems with the proposed modifications to the automatic rod control system will still prevent a reactor trip during the design load perturbations. Furthermore, the automatic rod control system does not provide any required automatic protective function nor is it utilized as a credited mitigating system for preventing radiological releases (dose consequences) for any UFSAR accident analysis. The operator can remove the system from service at any time by placing the control switch into the manual position, preventing any further rod motion commands from the systems from being implemented. Therefore, there is no impact on the consequences of an accident.

Removal of inputs to the automatic rod control system will reduce the probability of inadvertent rod stepping due to normal process swings and will eliminate unwarranted automatic rod motion from a failed N-44 channel. As noted in Question 3, the automatic control systems will continue to prevent a reactor trip for the design load perturbations (10% step change and 5% per minute ramp change). Furthermore, the automatic rod control system is not a credited mitigating system to prevent radiological releases (or dose consequences) due to any SSC malfunction. Therefore, the proposed activity doesn't result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety.

It is recognized in the UFSAR that the control rods may be manipulated either manually or automatically (insertion only in automatic). In addition, the power mismatch switch may be in either operate or defeat and by plant procedure, the pressure compensation switch is already maintained in the defeat position. As such, the changes are bounded by the existing design bases and analyses and will not introduce any new type of accident. As mentioned previously, the changes to the automatic rod control system do not increase the possibility of a malfunction within the system, but rather will reduce the potential for spurious rod stepping from normal plant system fluctuations and will eliminate automatic rod motion due to a failure in the N-44 channel. There are no new failure modes or effects and, as such, no new result from equipment malfunctions.

The design bases and safety analyses do not credit the automatic rod control system in any manner. Therefore, design basis limits for fission product barriers are not affected.

As mentioned previously, the design bases and safety analyses do not credit the automatic rod control system in any manner. While the configuration/response of the system is affected, there is no impact on evaluation methodologies used to develop the design bases or safety analyses.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 256394:

Description:

This evaluation pertains to changes to the Technical Requirements Manual (TRM) section TR 4.4.1 to allow a temporary increase from 184 days to 330 days for the turbine valve test frequency.

Summary of Evaluation:

Turbine overspeed event is not an accident evaluated in Chapter 15 of the UFSAR. The UFSAR does discuss this event in Chapter 3.5. The recommend change does not increase the likelihood of the turbine trip, only the probability of an overspeed following a turbine trip. The probability of overspeed event after a turbine trip is dependent on the likelihood of the failure of the turbine valves to close. The evaluation demonstrates that while the probability of an overspeed event is increased by the frequency change, it is still within the applicable NRC acceptance criteria. Thus, the proposed activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

As stated above, the evaluation provides the technical justification that this TRM change will not significantly increase the likelihood of occurrence of a malfunction of the turbine valves ability to close. Thus there is not a significant increase in the likelihood of turbine missile ejection that could cause a malfunction of equipment important to safety. Thus, the proposed activity will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

There is no change to the consequences to any accident because the probability of a turbine overspeed event remains acceptably low. Therefore, there is no change to the consequences of any accident previously evaluated in the UFSAR. There is no increase in the consequences of any malfunction that will result from extending the turbine valve test frequency. This conclusion is based on the determination that the likelihood of a destructive turbine ovespeed event has not been significantly increased and remains acceptably low, such that turbine overspeed events are considered to be precluded.

The probability of a turbine ovespeed resulting in missile ejection remains acceptably low. No physical changes are made to the plant or to how it is operated. Therefore, no different types of accident initiating events or event sequences are being created by this change. Thus the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

Extending the frequency of testing of the turbine valves during the current operating cycle does not affect any fission product barrier design basis limits. The accident analysis and safety system functions remain as described in the UFSAR. Therefore, the proposed activity will not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. The TRM change does not involve a change to any method of evaluation. The technical justification used for determining the acceptability of the proposed change is consistent with the current method of evaluation described in the UFSAR.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 263051:

Description:

This evaluation pertains to revised main steam line break (MSLB) analyses that demonstrate operation of HBRSEP, Unit No. 2, at power levels up to 2339 MWt during Operating Cycle 25 will comply with fuel design limits.

Summary of Evaluation:

There are no changes to plant SSCs or operating limits as a result of the proposed activity. The dose consequences analyses of record for Alternative Source Term (AST) are bounding for the correction to the MSLB analyses. Therefore, no change occurs to the consequences of the MSLB accident or the malfunction of a SSC important to safety during MSLB. The revised MSLB analysis demonstrated that the requirements and acceptance criteria defined in the UFSAR are satisfied. Therefore, the MSLB changes continue to support the plant licensing basis.

The proposed change involves analysis only and therefore has no impact on the frequency of MSLB, likelihood or consequences of the malfunction of SSCs, does not introduce an accident of a different type or result in a malfunction of an SSC that could lead to an accident of a different type. The proposed change does not involve new analysis methodology and does not involve a test or experiment.

The requirements and acceptance criteria defined in the UFSAR for Chapter 15 events are satisfied. The MSLB analyses continue to demonstrate that fuel failures do not occur. The correction does not impact the MSLB steam release that is an input to the dose analysis. Therefore the revision does not impact the current dose analysis.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).