10 CFR 50.59(d)(2)

Progress Energy

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
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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, 2008, and April 1, 2010. 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: Mr. L. A. Reyes, NRC, Region II
Mr. T. Orf, NRC Project Manager, NRR
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SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS FOR THE H. B. ROBINSON STEAM ELECTRIC PLANT (HBRSEP), UNIT NO. 2

Evaluations performed for changes made in accordance with 10 CFR 50.59 during the time period of April 1, 2008, to April 1, 2010:

Evaluation No. 263051:

Description:

This evaluation pertains to an update to the Main Steam Line Break (MSLB) analysis in the Updated Final Safety Analysis Report (UFSAR). The update was required when an error was discovered in the fuel centerline melt (FCM) analysis section of the MSLB analysis for Operating Cycle 25. In the FCM analysis, fuel rod specific FCM acceptance criteria were used to remove excessive conservatism from three End of Cycle (EOC) MSLB cases. The cases were:

- Hot Full Power (HFP) with Offsite Power Available
- Hot Zero Power (HZP) with Offsite Power Available (221 seconds case)
- HZP with Offsite Power Available (52 seconds case)

Errors occurred in the first two cases when the FCM results were incorrectly compared against acceptance criteria for individual rod types. The comparison was performed correctly for the HZP with Offsite Power Available (52 seconds) case.

Obtaining FCM results that satisfied the acceptance criteria required modifying several aspects of the Cycle 25 analysis. The changes were:

- Using the actual EOC burnup for Cycle 24 rather than the burnup window specified for Cycle 24 in the Cycle 25 Safety Analysis Report
- Reanalyze the ANF-RELAP transient for HFP with Offsite Power Available using a different scram worth to remove margin between the bounding core reactivity values and the Cycle 25 specific reactivity values
- Recalculating FCM results for the two cases in question and comparing the results to the applicable limits.

The results are summarized as follows:

- The FCM and minimum departure from nucleate boiling ratio (MDNBR) results were acceptable
- The ANF-RELAP point-kinetics reactivity models were demonstrated to be conservative relative to actual Cycle 25 design values
- The MSLB case with the minimum margin for either FCM or MDNBR limits changed to the HZP with Offsite Power Available case based on the FCM results

- The reported results for MDNBR remained unchanged because the inputs improved with respect to calculating MDNBR and the previous Cycle 25 value were therefore conservative.

The impacts on the UFSAR description of the MSLB included:

- Revised narrative in Section 15.1.5 to change the limiting event to HZP with Offsite Power Available
- Revision of UFSAR Tables and Figures to present the transient data for the limiting case.

The limiting MDNBR result reported in the UFSAR did not change. The UFSAR description of the FCM analysis for MSLB analysis does not include the limiting values and acceptance criteria and therefore did not change. The corrections did not impact operating limits, so there were no changes to plant configuration or plant procedures.

Summary of Evaluation:

The safety analyses for the MSLB were revised to correct an error. The corrections resulted in a change in the limiting MSLB case and required changes to the UFSAR to describe the limiting MSLB analysis. The re-analysis and evaluation confirmed that all applicable design basis limits for fission product barrier design criteria continued to be met for each MSLB case.

There were no changes to plant systems, structures or components (SSCs) or operating limits as a result of the changes.

The dose consequence analyses of record were bounding for the correction to the MSLB analyses. Therefore, no change occurred to the consequences of the MSLB accident.

The revised MSLB analysis demonstrated that the requirements and acceptance criteria defined in the UFSAR were satisfied. Therefore, the MSLB analysis changes continued to support the licensing basis.

The change involved analysis only and therefore has no impact on the frequency of a MSLB or likelihood of the malfunction of SSCs. The change did 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 change did not involve new analysis methodology and did not involve a test or experiment.

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

Evaluation No. 275712:

Description:

This evaluation pertains to revisions to the end-of-life (EOL) moderator temperature coefficient (MTC) operating and surveillance limits. The MTC operating limit is a design input to the safety analyses. Revision of the EOL MTC operating limit requires revision of the surveillance limits, which ensure that the operating limit is not exceeded. These limits are provided in the Core Operating Limits Report (COLR). The COLR was revised to incorporate the revised MTC design and surveillance limits. The COLR change also included an editorial change to clarify the MTC limit that is applicable at 50% power.

The revised analyses also supported operation with an increased unborated volume in the Safety Injection (SI) lines. The analyses were performed using the S-RELAP5 methodology, which has been reviewed and approved by the NRC for use at HBRSEP, Unit No. 2.

The revised analyses resulted in changes to the UFSAR for the following Chapter 15 events:

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

The UFSAR changes included revised results for comparison with Specified Acceptable Fuel Design Limits (SAFDLs) listed in the UFSAR or fuel failure acceptance criteria. The acceptance criteria are specified in terms of minimum departure from nucleate boiling ratio (MDNBR) and fuel centerline melt (FCM). The results of the analysis for the revised MTC were within the acceptance limits. The UFSAR revision also included deletion of a statement that the SI lines up to the final isolation valve remain filled with water borated to the refueling water storage tank boron concentration. This deletion did not change the plant procedures for filling the lines or testing the associated components. The deletion recognized that other plant conditions could dilute the boron concentration in these lines. Plant safety analyses conservatively bound the potential plant conditions by assuming the applicable volumes have unborated water.

The impact of the increased negative MTC at end of cycle was evaluated for impacts to the mass and energy release data that is used in containment pressure and temperature analysis. The change was found to have no impact on the mass and energy release rates and thus no impact on accident temperature and pressure inside of containment. The evaluation of the change included an evaluation of potential impacts on the fission product barriers and no adverse affects were identified.

Summary of Evaluation:

The revised analyses demonstrated that the affected events continue to meet their acceptance criteria. Therefore, the consequences of an accident or a malfunction are unaffected by the proposed changes. There are no new accidents or new SSC failure mechanisms introduced by the proposed activities. There are no increases in the likelihood of an SSC malfunction because the proposed activity involves no changes to SCCs. The EOL MTC operating and surveillance limits and assumed unborated volume in the SI system provide boundary conditions for system responses; these inputs do not initiate or contribute to the possibility of an accident including any previously evaluated in the UFSAR. The analyses performed in support of the MTC limit change evaluate the system response and consequences of those events; the analyses do not affect the frequency of occurrence of an accident previously evaluated in the UFSAR. The analyses were performed using S-RELAP5, which has been reviewed and approved by the NRC for HBRSEP, Unit No. 2, as documented in the Core Operating Limits Report (COLR).

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. 292186</u>:

Description:

This evaluation pertains to the addition of a Control of Heavy Loads summary to the UFSAR. Station procedures were also revised to include the restrictions imposed by a new reactor vessel head drop evaluation. Appropriate restrictions were added based on the height of head movements. The appropriate notes, steps, and cautions were added to ensure compliance with the limitations.

Nuclear Energy Institute (NEI) Initiative 08-05 on the control of heavy loads was adopted in September 2007. It contains several requirements including the following: "In your next FSAR update, provide a summary description of your basis for conducting safe heavy load movements, including commitments to safe load paths, load handling procedures, training of crane operators, use of special lifting devices, use of slings, crane design, and inspection, testing, and maintenance of the crane. If the safety basis includes reliance on a load drop analysis, then that fact should be included in the summary description within the FSAR."

This UFSAR revision replaced the incorporation by reference of the control of heavy loads with a summary description as required by the NEI Initiative. The revision also incorporates references to a reactor vessel head drop evaluation based on NEI 08-05 and NRC accepted methodologies. This evaluation did not exist prior to this UFSAR update.

There are no physical changes being made to any SSCs as a result of this activity. The proposed activity does not adversely change the manner in which heavy loads are handled. The addition to the UFSAR of a section pertaining to the control of heavy loads, where no explicitly described function existed prior, enhances the ability to safely move heavy loads.

The addition to the UFSAR of a section on the control of heavy loads includes a reference to a reactor vessel head drop evaluation. The evaluation of reactor vessel head drop required changes to procedures involved with the reactor vessel head lift and cavity filling/draining. The procedure changes introduced hold points during the reactor vessel head removal and replacement and the cavity filling/draining such that the bounding conditions of the reactor vessel head drop evaluation will be maintained. The changes do not impact the water levels in the cavity required for shielding. The hold points do not adversely alter or impact the existing means of performing or controlling the design functions for the SSC's involved. No operating parameters or set points are changed.

Summary of Evaluation:

Other than incorporating improved procedural controls during a vessel head lift to reduce the potential for consequences of an unexpected load drop, there are no changes to plant design or procedures. Therefore, there is no negative impact on the probability or consequences of an accident or malfunction and no new accidents or malfunctions created. The requirements for the handling of heavy loads have not changed from previously docketed information with the exception of including a load drop evaluation and the procedural controls incorporated to address the load drop evaluation. The reactor head drop evaluation ensures reactor core cooling is maintained after an unanticipated vessel head drop. The methodology and acceptance criteria used in this evaluation have been performed in accordance with NEI 08-05. In a September 5, 2008, letter the NRC provided their acceptance of the methodology and acceptance criteria of NEI 08-05, with the exception of the strain based acceptance criteria for coolant containing components. This NRC letter allows for the employment of this analysis methodology within the 10CFR50.59 process. The HBRSEP, Unit No. 2, evaluation does not use the strain based acceptance criteria and hence is bounded by the NRC SER.

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. 302857:

Description:

This evaluation pertains to a new core loading pattern and associated analyses in support of reactor core reload for Cycle 26 operation. The major changes included a new reload batch neutronic design and associated safety analysis, new core loading pattern, updates to the Updated Final Safety Analysis Report (UFSAR), and associated changes to plant procedures.

The Cycle 26 core contains 8 ROB-19 High Thermal Performance (HTP) assemblies, 8 ROB-20 HTP assemblies, 20 ROB2-24 HTP assemblies, 53 ROB2-25 HTP assemblies, 11 ROB-19 Part Length Shield Assemblies (PLSA), 1 ROB-13 PLSA and 56 fresh ROB2-26 HTP assemblies. (The "ROB" and "ROB2" are designators for the individual fuel regions). The 8 ROB-19, 8 ROB -20 HTP, and 1 ROB-13 assemblies are re-inserts that had been previously discharged to the spent fuel pool. The PLSAs are used to reduce the fast neutron fluence reaching the pressure vessel wall. The Cycle 26 assemblies utilize the debris resistant FUELGUARD™ (trademark of fuel manufacturer AREVA NP) lower tie plate.

The mechanical design of the ROB2-26 (Region 29) fuel is similar to that of the ROB2-25 (Region 28) fuel loaded in Cycle 25. The fuel assemblies used in the Cycle 26 core have been designed in accordance with AREVA NRC-approved design criteria (EMF-92-116). Minor changes in the mechanical design do not adversely affect the design function of the fuel assembly.

Several fabrication surveillances were conducted by Progress Energy to ensure the fuel was fabricated as designed. The surveillance campaign concluded that the fuel is acceptable for use at HBRSEP, Unit No. 2.

The core loading pattern was developed in accordance with applicable procedures. The functional requirements for the Cycle 26 reload design were established, including the core reload design goals. The design goals provide additional margin beyond that required by the Technical Specifications and Core Operating Limits Report for Cycle 26. The functional requirements are evaluated with the computer codes SIMULATE and MICROBURN during design development. The PRISM code is used for the neutronics input to the safety analysis.

The reload analyses support Cycle 26 operation at a nominal core power of 2339 Megawatts-thermal (MWt) for up to 522 effective full-power days (EFPD). The Cycle 26 safety analyses are based on the actual Cycle 25 shutdown energy. It was concluded that Cycle 26 design meets the functional requirements based on the PRISM results.

The analyses of record (AOR) for several non-Loss of Coolant Accident (non-LOCA) events were changed by the reload analyses. The changes are:

- The boron dilution events were reanalyzed. The reanalysis corrected errors discovered in previous analysis and included cycle specific reactivity parameters. The reanalysis resulted in changes to the shutdown margin required in the Core Operating Limits Report (COLR).

- Non-conservatism was found in the shutdown margin used to prevent criticality from uncontrolled rod withdrawal with less than two reactor coolant pumps in operation. This did not require reanalysis of the event, but increased the shutdown margin in the COLR for the specific operating condition.

- Hot channel analyses for Minimum Departure from Nucleate Boiling Ratio (MDNBR) and Fuel Centerline Melt (FCM) were performed to account for Cycle 26 power shapes.

For Cycle 26, the methodology for Large Break Loss of Coolant Accident was changed to Realistic Large Break LOCA (RLBLOCA). The methodology was previously approved by the NRC for use by HBRSEP, Unit No. 2, but cycle neutronic design had prevented earlier adoption of the RLBLOCA method until Cycle 26. The use of RLBLOCA methods resulted in a lowering of peak clad temperature (PCT) for the event.

The analyses performed demonstrate that operation at up to 2339 MWt during Cycle 26 complies with the Technical Specifications. The requirements and acceptance criteria defined in the UFSAR for Chapter 15 events were also satisfied. Events that did not previously predict fuel failures continue to demonstrate no fuel failures. For those Chapter 15 events that result in predicted fuel failure, the amount of fuel failure predicted for Cycle 26 remains within the assumptions of the Alternative Source Term (AST) dose analyses of record.

The Cycle 26 changes to the UFSAR include:

- Discussion added to Section 4.3 concerning analysis of the one ROB-13 PLSA identified as X45.
- The rod design was changed in the Cycle 25 reload to include a new lower end cap; this was not included in the Cycle 25 description. The fuel drawing showing fuel assembly length was revised to incorporate new AREVA drawing standards and the round off for fuel pin length is affected. The drawing change had no impact on fuel design.
- The limiting axial power distribution for Departure from Nucleate Boiling (DNB) events in Figure 15.0.3-1
- MDNBR and FCM results changed based on cycle specific results.
- In the Main Steamline Break event (UFSAR 15.1.5), Uncontrolled Rod Withdrawal (UFSAR 15.4.2), and Dropped Rod/Bank (UFSAR 15.4.3.3), the limiting case for MDNBR and FCM changed. This affected the applicable UFSAR tables and figures.
- Boron concentration and time-to-criticality results for the Decrease in Boron Concentration event (15.4.6).
- The misload analysis was revised to present specific information on Cycle 26 design and deleted information that is only applicable to the misloads involving the center assembly in Cycle 25.
- Revised Large Break Loss of Coolant Accident performed with AREVA Realistic LBLOCA analysis method. This resulted in a complete replacement of the LBLOCA UFSAR Section.

Updates to the Core Operating Limits Report (COLR) for Cycle 26 include revising the 300 parts per million boron (ppm B) and 60 ppm B MTC Surveillance Limits, V(z) curve, k(z) curve and other cycle specific data. The changes to these values do not affect how the respective surveillances are performed. In addition, the required shutdown margin (SDM) specified for Mode 3 was increased to a minimum of 1.3% $\Delta k/k$ and the Mode 5 SDM remained 4% $\Delta k/k$, but was no longer constrained to use 1950 ppm. These SDM changes have no adverse effects on safety. Various operating procedure changes were also made to implement the Cycle 26 analyses.

Summary of Evaluation:

The plant was analyzed for Cycle 26 operation with a reload batch of 56 fresh natural uranium axial blanket (NUAB) fuel assemblies. The ROB2-26 (Region 29) NUAB fuel assemblies contain Gadolinia-bearing fuel rods. The 56 new assemblies contain High Thermal Performance (HTP) spacers, Intermediate Flow Mixer (IFM) grids, and FUELGUARD™ debris-resistant lower tie plates. Cycle 26 is the seventeenth HBRSEP, Unit No. 2, core reload of NUAB assemblies and the seventeenth successive reload containing Gadolinia-bearing fuel.

The safety analyses support Cycle 26 operation at a nominal core power level of 2339 MWt for up to 522 effective full-power days (EFPD). The Cycle 26 safety analyses are based on the actual Cycle 25 shutdown energy.

The following areas were evaluated to support Cycle 26: 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 26 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 26.

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) was required. The effect of DNB propagation on fuel failure is reflected in the results for Cycle 26.

The analysis of record Alternative Source Term (AST) LOCA doses are bounding for Cycle 26. 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 26.

Safety analyses were reviewed with respect to Cycle 26 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 26. Cycle 26 was the first use of Realistic LOCA as the analysis of record for Large Break LOCA. The revised analyses and evaluation confirmed that the applicable acceptance criteria continue to be met for each event.

As described in the above discussion of the acceptability of the analysis results, implementation of the Cycle 26 core design and supporting safety analyses demonstrated that the requirements and acceptance criteria defined in the UFSAR are satisfied for Cycle 26 operation. Therefore, the Cycle 26 reload design, with regard to the safety analysis, will continue to meet the plant licensing basis.

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. 387078:

Description:

This evaluation pertains to a new procedure titled, "Acoustic Emission Inspection of Reactor Head Lifting Rig and Internals Lifting Rig." This procedure is for the performance of a Non-Destructive Examination (NDE) inspection of the Reactor Head and Internals Lifting Rigs to satisfy the requirements contained in Generic Letter 81-07, "Control of Heavy Loads" and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The lift rigs will be monitored under service loading by acoustic emission defect detection. HBRSEP, Unit No. 2, has a commitment in accordance with NUREG-0612 to inspect the Vessel Head Lift Rig, and the Internals Lift Rig every 5 years in accordance with ANSI 14.6-1978. ANSI 14.6-1978 describes the various inspection techniques such as visual, liquid penetrant, and magnetic particle testing. There is no specific discussion with respect to the acoustic emission technique. The intent of the ANSI 14.6 and NUREG-0612 guideline is to detect potential cracks and or flaws rendering the special lifting equipment not operable. The technique associated with a type of NDE inspection is being changed, however the intent associated with detecting flaws and or cracks is not being altered. In a Safety Evaluation Report for the Sequoyah Nuclear Plant, the NRC concluded that "the acoustic emission monitoring of the reactor vessel head and reactor internals lifting rigs provides adequate assurance for safe operation." Vendor developed software is used to evaluate the acoustic emission data and correlate that information to a location of the potential flaw. Upon detection of a potential flaw, a volumetric ultrasonic inspection or a combination of penetrant or magnetic particle testing techniques can be employed to characterize the potential flaw. UFSAR Section 9.1.5.5 will be revised to reflect the alternative method used to inspect the vessel head lift rig and internals lift rig.

Acoustic emission (AE) depends on the detection of ultrasonic noises generated by discontinuities or flaws when under load, even at loads well below those of engineering stress limits. AE can detect the presence of these flaws by their emitting acoustical vibrations, and with several transducers and appropriate supporting equipment, position the source of the emissions. One of the advantages of AE over other nondestructive inspection techniques used is that it does provide a warning of a flaw growing. Because of the nature of the structural materials that make up the lifting rig, their overall performance is regarded as ductile in comparison to truly brittle materials. The ductility allows for the flaw to grow to a critical size before sudden catastrophic failure. The growth of the flaw generates ultrasonic sounds that give warning of the active flaw. With structural steels at room temperature, there is adequate toughness such that AE will provide a warning to allow the removal of the load from the component before failure occurs. AE monitoring usually starts with the component not under load, and it is monitored as the load increases. If the flaw is active, it makes its presence known at low percentages of full load, so the load may be removed and the cause of the AE determined before reapplying the load.

The AE method detects the presence of active flaws under load, and with software processing of the data generated, can locate the source of the acoustic emission. It cannot measure the size of a flaw, or characterize the flaw. After the flaw is detected and located by AE, a measurement and characterization of the flaw can be conducted by other NDE techniques. If there are no AE emitting flaws in a structure under load, no further NDE of the lift rig is required. Even if a structure has no flaws, but is overloaded such that there is permanent deformation, AE will detect such conditions, and provide a warning of the overload. AE provides a much higher assurance than surface or visual examinations, which can only detect and provide a measure of flaws on the surface.

Summary of Evaluation:

Based on the above discussion, it can be concluded that the AE technique is an acceptable method of examination with respect to detecting a flaw. It is the intent of the NDE inspection to detect the flaws and determine their locations. If a flaw is detected, additional techniques would be used to characterize the size and depth of the potential flaw. These additional techniques are part of the overall process associated with ensuring that the lift rigs would perform safely. Therefore, employing the AE technique will not increase the frequency of failure of the associated lift rigs, hence will not increase the frequency of occurrence of the accidents that could be caused by a load drop.

In the Sequoyah SER, the NRC Staff concluded that the acoustic emission monitoring technique of the reactor vessel head and reactor internals lifting rigs provides adequate assurance for safe operation. The HBRSEP, Unit No. 2, process is equivalent to the inspection process used at Sequoyah. Therefore, the SER for Sequoyah related to this inspection technique is applicable to HBRSEP, Unit No. 2. The intent of the requirements in ANSI 14.6-1978 and NUREG-0612 is to detect potential flaws that could render the special lifting equipment inoperable. The technique associated with the type of NDE inspection is being changed, however the requirement to detect flaws is not being altered. The frequency of inspection is based on formation and propagation of a crack. HBRSEP, Unit No. 2, will maintain the 5 year inspection frequency previously approved by the NRC for the NDE inspection techniques described in N14.6-1978.

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).