Donald C. Cook Nuclear Plant
Report of Changes, Tests, Experiments Pursuant to 10 CFR 50.59(d)(2)

As required by 10 CFR 50.59(d)(2), the following report contains brief descriptions of changes made to the facility and/or associated documentation, and summaries of the associated 50.59 evaluations.

SS-SE-2017-0411-02
D.C. Cook Unit 2 Up-flow Conversion

ACTIVITY DESCRIPTION:

This activity implemented an "up-flow conversion" modification at D.C. Cook Unit 2. In summary, the activity consists of a flow conversion in the baffle-barrel region from a "down-flow" design to an "up-flow" design. The change is achieved by plugging the existing core barrel flow holes and machining new flow holes into the top former plate. The effects of this modification upon safety analysis inputs include:

- higher core bypass flow
- lower flow through the core
- a reduction in differential pressure across the core
- a reduction in the differential pressure across the baffle joints

SUMMARY OF THE EVALUATION:

The analyses governed by 10 CFR 50.59 process, performed to assess the effect of the up-flow conversion modification of the D.C. Cook Unit 2 plant upon the nuclear unit's design and licensing basis, concluded that the modified unit meets all applicable design requirements. Moreover, the current core and axial offset limits remain applicable and the Departure from Nucleate Boiling Ratio (DNBR) design basis continues to be met for the DNB transients and the minimum DNBR and peak linear heat generation rate limits are met. The Peak Cladding Temperature (PCT) 10 CFR 50.46 limits also continue to be met. The conclusions of the Updated Final Safety Analysis Report (UFSAR) remain valid for all the transient and accident analysis.

None of the analysis and investigation of possible failure modes and effects due to the physical change indicated that a new credible accident or malfunction would be created as a result of this activity. This activity is not an accident or malfunction initiator. Therefore, the existing accident initiating conditions, scenarios, and mitigating mechanisms are unchanged.

Impacts upon some safety analysis inputs necessitated that those safety analyses be re-run to confirm that applicable design limits and acceptance criteria continue to be met. When safety analyses need to be re-run to confirm that the design limits and acceptance criteria continue to be met, it is required to be addressed in a 10 CFR 50.59 Evaluation.

The Small and Large Break Loss of Coolant Accident (LOCA) analyses, the Rod Withdrawal from Subcritical (RWFS) analysis, and the Hot Zero Power Steamline Break (SLB) analysis, as well as a number of supporting safety analyses needed to be rerun to demonstrate that all required safety functions and design requirements are met. The revised analyses confirmed that design limits and acceptance criteria continue to be met.

SS-SE-2019-0328-00
D.C. Cook Unit 1 Up-flow Conversion

Activity Description:

This activity implemented an "up-flow conversion" modification at D.C. Cook Unit 1. In summary, the activity consists of flow conversion in the baffle-barrel region from a "down-flow" design to an "up-flow" design. The change is achieved by plugging the existing core barrel flow holes and machining new flow holes into the top former plate. The effects of this modification upon safety analysis inputs include:

- higher bypass flow
- lower flow through the core
- a reduction in differential pressure across the core
- a reduction in the differential pressure across the baffle joints

Summary of the Evaluation:

The analyses governed by 10 CFR 50.59 process, performed to assess the effect of the upflow conversion modification of the D.C. Cook Unit 1 plant upon the nuclear unit's design and licensing basis concluded that the modified unit meets all applicable design requirements. Moreover, the current core and axial offset limits remain applicable and the DNBR design basis continues to be met for the DNB transients and the minimum DNBR and peak linear heat generation rate limits are met. The PCT 10 CFR 50.46 limits also continue to be met. The conclusions of the UFSAR remain valid for all the transient and accident analysis.

None of the analysis and investigation of possible failure modes and effects due to the physical change indicated that a new credible accident or malfunction would be created as a result of this activity. This activity is not an accident or malfunction initiator. Therefore, the existing accident initiating conditions, scenarios, and mitigating mechanisms are unchanged.

Impacts upon some safety analysis inputs necessitated that those safety analyses be re-run to confirm that applicable design limits and acceptance criteria continue to be met. When safety analyses need to be re-run to confirm that the design limits and acceptance criteria continue to be met it is required to be addressed in a 10 CFR 50.59 Evaluation.

The Small and Large Break LOCA analyses, the RWFS analysis, and the Hot Zero Power SLB analyses (inputs were no longer bounded), as well as a number of supporting safety analyses needed to be rerun to demonstrate that all required safety functions and design requirements are met. The revised analyses confirmed that design limits and acceptance criteria continue to be met.

SS-SE-2018-0006-00
Changes to Decay Heat Assumption in Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology

Activity Description:

This change involved revising the Unit 2 steam generator tube rupture (SGTR) margin to overfill (MTO) analysis. The change was completed to address potentially non-conservative treatment of decay heat uncertainties as identified in Westinghouse NSAL 07-11, Decay Heat Assumption in Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology. The alteration involved changes to certain input parameters used in the analysis and a change to the approved method used for evaluation of SGTR MTO. Input parameters were changed for reactor thermal power level, auxiliary feedwater pump start delay time, steam generator Power Operated Relief Valve steam flow, and auxiliary feedwater purge volume. In addition, the treatment of decay heat uncertainties was changed from the method described in WCAP-10698-P-A, SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill. WCAP-10698-P-A specifies that maximum decay heat (ANS 1970+20%) is to be assumed when performing SGTR MTO analyses, the proposed change assumed minimum decay heat (ANS 1979-2sigma) to address the issue identified in Westinghouse NSAL 07-11.

Summary of the Evaluation:

The SGTR MTO analysis is performed to verify that the ruptured steam generator does not overfill. Ensuring the ruptured steam generator does not overfill validates that the assumptions of the SGTR dose analysis remain bounding. Although the MTO was reduced, the new analysis demonstrated that the ruptured steam generator does not overfill considering the revised input parameters and the change to the method of evaluation. Because the change to the decay heat assumed (an element in the method of evaluation) produces results closer to the applicable limit, the change in method is considered conservative and not a departure from the method of evaluation formerly applied in the safety analysis. Since steam generator overfill does not occur in the Analysis of Record or the revised MTO analysis, the proposed activity does not result in any change in the consequences of an accident or malfunction previously evaluated in the UFSAR and prior Nuclear Regulatory Commission (NRC) approval was not required.

SS-SE-2018-0087-00
Pre-Critical Alignment and Hookup for DRWM, Low Power Physics Tests with Dynamic Rod Worth Measurement

Activity Description:

This activity implemented low power physics testing (LPPT) using the Westinghouse RhoPRO technology. This entails using the Intermediate Range Nuclear Instrumentation (IR) for the Dynamic Rod Worth Measurement (DRWM) technique. With the exception of using the IR detectors as input, the methodology described in the approved topical report for DRWM remains unchanged. The scope of this change also considered alterations to the procedures governing the setup and execution of DRWM that accommodate use of the IR detectors in lieu of the Power Range Nuclear Instrumentation (PR) detector. The previous DRWM technique removed a PR detector from service in order to route its neutron flux signals to a reactivity computer. The RhoPRO DRWM technique uses the neutron flux signal from existing spare IR analog outputs with the IR channels remaining in service, leaving their functions unaffected.

Summary of the Evaluation:

The proposed change (i.e., using the IR inputs) was compared to the existing use of PR inputs. Westinghouse benchmarked the new methodology against the existing methodology at multiple sites. The measurements using each type of detector were performed in parallel under normal LPPT program following a reload startup, so the results represent the same set of plant conditions and were readily compared. The result of this benchmarking demonstrated that the results using the IR inputs are essentially the same as results using the PR inputs. Because the revised method produces results that are essentially the same as those produced by the NRC approved method, the change does not meet the definition of a Departure from a Method of Evaluation Described in the Final Safety Analysis Report (as Updated) and prior NRC approval is not required.