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L-2019-044
10 CFR 50.59(d)

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Re: St. Lucie Unit 2
Docket No. 50-389
Report of 10 CFR 50.59 Plant Changes

Pursuant to 10 CFR 50.59(d)(2), the attached report contains a brief description of any changes, tests and experiments, including a summary of the evaluation of each, which were made on Unit 2 during the period of Amendment No. 25 (March 23, 2017 through October 3, 2018) dated April 2019. This submittal correlates with the information included in Amendment 25 of the Updated Final Safety Analysis Report submitted under a separate cover letter.

Should you have any questions regarding this submittal, please contact Mr. Michael J. Snyder, Licensing Manager, at 772-467-7036.

Sincerely,

Michael J. Snyder

Michael J. Snyder
Licensing Manager
St. Lucie Plant

MJS/rcs

Enclosure

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, St. Lucie Plant
USNRC Senior Resident Inspector, St. Lucie Plant

ST. LUCIE UNIT 2
DOCKET NUMBER 50-389
CHANGES, TESTS AND EXPERIMENTS
MADE AS ALLOWED BY 10 CFR 50.59
FOR THE PERIOD OF
MARCH 23, 2017 THROUGH OCTOBER 3, 2018

INTRODUCTION

This report is submitted in accordance with 10 CFR 50.59 (d)(2), which requires that:

i) changes in the facility as described in the SAR;
ii) changes in procedures as described in the SAR; and
iii) tests and experiments not described in the SAR

that are conducted without prior Commission approval be reported to the Commission in accordance with 10 CFR 50.90 and 50.4. This report is intended to meet these requirements for Amendment 25 addressing the period of March 23, 2017 through October 3, 2018.

This report is typically divided into three (3) sections:

First, changes to the facility as described in the Updated Final Safety Analysis Report (UFSAR) performed by a Permanent Modification are addressed.

Second, changes to the facility / procedures as described in the UFSAR, or tests/experiments not described in the UFSAR, which are not performed by a Permanent Modification, are addressed.

Third, a summary of any Core Reload 10 CFR 50.59 evaluation is addressed.

Sections 1, 2 and 3 summarize specific 10 CFR 50.59 evaluations that evaluated the specific change(s). Each of these 10 CFR 50.59 evaluations concluded that the change does not require a change to the plant technical specifications, and prior NRC approval is not required.

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SECTION 1

PERMANENT MODIFICATIONS

EC 289282, REVISION 0

UNIT 2 WIRELESS INFRASTRUCTURE PROJECT

SUMMARY:

EC 289282 adds to St. Lucie's wireless local area network (LAN) by installing a wireless network communications system within the Unit 2 Reactor Auxiliary Building (RAB), Turbine Building (TB), Fuel Handling Building (FHB), Diesel Generator Building (DGB), Component Cooling Water (CCW) Building, and Intake Structure. The wireless infrastructure allows in-plant access to nuclear business applications such as e-mail, Nuclear Asset Management Suite (NAMS) and Electronic Work Package (EWP).

Creating the new wireless infrastructure involves installation of data racks (consisting of Ethernet switches, power supplies and patch panels), raceway, cables, and wireless access points in various locations. Existing non-safety related power or lighting panels are used to power this equipment.

The expansion of the existing wireless system will increase the level of Electromagnetic interference (EMI) and radio-frequency interference (RFI) in the areas where the data racks and wireless access points are installed. The increase in electrical noise in the RAB, TB, FHB, DGB, CCW Building, and Intake Structure has the potential to have an adverse impact on the environment and design functions of SSCs installed in the RAB, TB, FHB, DGB, CCW Building, and Intake Structure. All other aspects of EC 289282 screened out from further 10 CFR 50.59 evaluation. The 10 CFR 50.59 Evaluation applied the guidance of EPRI TR-102348 Revision 1 (NEI 01-01), "Guideline on Licensing Digital Upgrades."

EC 289282 documents an evaluation of the impact of Electromagnetic and Radio-Frequency Interference generated by the wireless infrastructure backbone components including the location of these components to surrounding components in the RAB, TB, FHB, DGB, CCW Building, and Intake Structure to demonstrate that there is adequate margin between the emission and equipment susceptibility levels such that the installation will not adversely impact EMI/RFI sensitive equipment installed within the RAB, TB, FHB, DGB, CCW Building, and Intake Structure. This evaluation is based upon the guidance in Appendix I of EPRI TR 102323 Revision 4 and maintains the margins recommended in Regulatory Guide 1.180 Revision 1. EC 289282 justifies selection of the equipment susceptibility levels for non-tested legacy equipment based on guidance provided in EPRI TR 102323 Revision 4. To ensure that the increase in electrical noise in the RAB does not impact SSCs within the Control Room, EC 289282 does not install wireless access points at EL 62' in the RAB.

The selection of wireless communication devices with sufficiently low emission levels coupled with the placement of these devices sufficiently far from EMI/RFI sensitive equipment is credited in EC 289282 to ensure electromagnetic compatibility with EMI/RFI sensitive equipment. There are no known or anticipated failure modes of the wireless infrastructure backbone components that could result in an increase in the EMI/RFI interference generated by the wireless equipment. The installation and testing plan in EC 289282 includes requirements that prior to energizing the access points, a physical verification be performed to ensure that the access points meet the required separation distance from EMI/RFI sensitive equipment.

As such, the UFSAR described design functions of equipment installed within the RAB, TB, FHB, DGB, CCW Building, and Intake Structure that could be susceptible to Electromagnetic and Radio-Frequency Interference are not adversely impacted by EC 289282.

Therefore, the increase in electrical noise within the RAB, TB, FHB, DGB, CCW Building, and Intake Structure does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR or more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

For the reasons discussed above, it was also concluded that the increase in electrical noise within the RAB, TB, FHB, DGB, CCW Building, and Intake Structure does not result in more than a minimal increase in the radiological consequences of a malfunction of an SSC important to safety or an accident previously evaluated in the UFSAR.

The evaluated lack of adverse impact also lead to the conclusion that the increase in electrical noise within the RAB, TB, FHB, DGB, CCW Building, and Intake Structure does not create a possibility for a malfunction of an SSC important to safety with a different result or an accident of a different type than any previously evaluated in UFSAR.

There are no numerical values in the UFSAR that are used directly in the determination of the integrity of the fission product barriers that are associated with the change in the electrical noise level in the RAB, TB, FHB, DGB, CCW Building, and Intake Structure environment. Therefore, the increase in electrical noise within the RAB, TB, FHB, DGB, CCW Building, and Intake Structure does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

There are no methods of evaluation described in the UFSAR impacted by the change.

Because the proposed change does not require a change to the technical specifications and does not meet any of the criteria in 10 CFR 50.59(c)(2), the change can be made without obtaining a license amendment pursuant to 10 CFR 50.90.

EC 291956, REVISION 0

REPLACE UNIT 2 PRESSURIZER LIQUID SPACE INSTRUMENT NOZZLE 016-2B

SUMMARY:

The modification replaces an Alloy 600 weld pad for the lower pressurizer instrument nozzle 016-2B with an Alloy 690 weld pad. All other components in the nozzle replacement are like-for-like.

Replacement of the Alloy 600 weld pad with an Alloy 690 weld pad will not increase the frequency of occurrence of an accident previously evaluated in the UFSAR. The change in Reference Temperature for Nil Ductility Transition (RTndt) for the pressurizer bottom head continues to be bounded by the Technical Specification Lowest Service Temperature.

This activity does not result in any increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. The weld material on the pressurizer will not cause any equipment malfunction. It also has no impact any radiological analysis.

Replacement of the Alloy 600 weld pad with an Alloy 690 weld pad will not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR. Failure of the RCS pressure boundary is considered in the UFSAR, and this change will not create any new failure types or cause a different result.

Replacement of the Alloy 600 weld pad with an Alloy 690 weld pad will not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. The ambient temperbead weld procedure qualification report (PQR) identifies a resulting base metal temperature change in RTndt of +5°F. This results in a change for the pressurizer lower head RTndt from +10 °F to +15 °F. Technical Specification Figure 3.4-2 shows the "lowest service temperature" (LST) as 160 °F. WCAP 16817-NP Rev 2 identifies the highest RTndt for the St. Lucie Unit 2 RCS is +60 °F (TS value – 100 °F). Since the revised RTndt of 15 °F for the PZR lower head is significantly below the lowest service temperature of 60 °F, adequate toughness exists for the pressurizer lower head and the design basis limit is not exceeded or altered.

Replacement of the Alloy 600 weld pad with an Alloy 690 weld pad does not utilize any different method of evaluation than those described in the UFSAR used in establishing the design bases or in the safety analyses.

Because the proposed change does not require a change to the technical specifications and does not meet any of the criteria in 10 CFR 50.59(c)(2), the change can be made without obtaining a license amendment pursuant to 10 CFR 50.90.

SECTION 2

10 CFR 50.59 EVALUATIONS

EC 290539, REVISION 0

UNIT 2 UFSAR CORRECTION FOR SECTION 15.6 DUE TO MODELING ERROR

SUMMARY:

EC 290539 corrects an error in the Control Room dose consequences analysis that includes a concurrent (accident induced) iodine spike. The error is related to the use of the improper release fraction and timing file associated with the concurrent iodine spikes, which is mainly for Letdown Line Rupture (LLR) and Steam Generator Tube Rupture (SGTR) Analyses for St. Lucie Unit 2. Due to the adverse impacts of the modeling error, LLR and SGTR Control Room (CR) dose consequences in PSL Unit 2 UFSAR (Tables 15.6.2-3 and 15.6.3-4) required an update.

The UFSAR corrections regarding the impact on the Letdown Line Rupture and SGTR concurrent iodine spike doses due to the modeling error do not introduce the possibility of a change in the frequency of an accident. The calculations error correction and resulting doses, which meet the acceptance criterion, are not an input to determining the frequency of occurrence of an accident.

These UFSAR corrections do not introduce the possibility of a change in the likelihood of a malfunction because the calculations error correction and resulting doses, which meet the acceptance criterion, are not an initiator of any malfunctions and no new failure modes are introduced.

The revised St. Lucie Unit 2 Letdown Line Rupture and SGTR radiological dose analyses continue to meet the regulatory radiological dose acceptance criterion of the control room dose 5 REM TEDE. The revised analyses show that the increase in dose due to the adverse impact of the modeling error is small, and resulted in an overall control room dose increase of 2.9% for the Letdown Line Rupture and 1.3% for the SGTR, which are < 4% of the margin to the limit. This increase is less than the 10% of the margin to the limit of NEI 96-07 Section 4.3.3 and the dose increase would be considered no more than a minimal increase.

Therefore, the UFSAR changes regarding the impact on the Letdown Line Rupture and SGTR concurrent iodine spike doses due to correcting the modeling error do not result in a more than a minimal increase in the radiological consequences of an accident previously evaluated in the UFSAR.

These UFSAR corrections do not introduce the possibility of a change in the radiological consequences of a malfunction because the correction of the calculations error and the resulting doses, which meet the acceptance criterion, are not an initiator of any malfunctions, and no new failure modes are introduced. The proposed UFSAR changes also do not exceed or alter a design basis limit for a fission product barrier as described in the UFSAR.

These UFSAR corrections do not introduce the possibility of a new accident because the error correction and resulting doses, which meet the acceptance criterion, are not an initiator of any accident and no new failure modes are introduced. It is for this same reason that these corrections do not do not introduce the possibility for a malfunction of an SSC with a different result.

This UFSAR changes regarding the impact on the Letdown Line Rupture and SGTR concurrent iodine spike doses due to the modeling error do not constitute a change in method of evaluation as defined in Section 3.4 of NEI 96-07. The revised analyses continue to use the same methodology as used in previous calculations based on approved methodology. Therefore, the UFSAR changes do not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Because the proposed change does not require a change to the technical specifications and does not meet any of the criteria in 10 CFR 50.59(c)(2), the change can be made without obtaining a license amendment pursuant to 10 CFR 50.90.

EC 291265, REVISION 0

UNIT 2 SPENT FUEL POOL IMPROVED OFFLOAD TIMING

SUMMARY:

EC 291265 updates the UFSAR to allow earlier core offloading times and to revise the thermal overshoot value. The peak local water temperature and time to boil information will additionally be updated based on the new offload times. A reduction in the offload start time is determined to adversely affect the peak local temperature and thermal overshoot described in UFSAR.

The time to begin core offload is not used to determine the frequency of occurrence of any accident. The offload beginning time is only an input to the Spent Fuel Pool (SFP) cooling analysis. Therefore, the change does not result in in the frequency of occurrence of an accident as described in the UFSAR.

Similarly, the beginning of offload time is not used to determine the likelihood of occurrence of a malfunction of an SSC important to safety; therefore, the change does not result in a change in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The proposed reduction in the core offload start time results in a small increase in the peak local temperature. The local water and fuel cladding temperatures continue to satisfy the acceptance criteria with no impact on any radiological consequences analysis. The proposed change also has a small impact on the thermal overshoot in the SFP temperature with the failure of one SFP cooling pump. The upper limit on the SFP temperature set during core offload will continue to ensure the SFP bulk temperature remains below the acceptance criteria. There is no impact on any radiological analysis. Therefore, there is no change in the radiological consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR. Similarly, there is no change in the radiological consequences of an accident previously evaluated in the UFSAR.

Regarding the possibility to create an accident of a different type than any previously evaluated in the UFSAR, the proposed change to the offload start time is only an input to the SFP cooling analysis, which continues to meet the analysis acceptance criteria, and there is no interaction with any SSC for creating a different type of accident. Therefore, there is no possibility for an accident of a different type than previously evaluated in the UFSAR to occur. Similarly and since this change does not contribute to the malfunction of any SSC, there is no possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR.

No limits are impacted as a result of the changes: therefore, there is no design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

The proposed change to the core offload start time is analyzed using the same method of evaluation currently described in the UFSAR.

Because the proposed change does not require a change to the technical specifications and does not meet any of the criteria in 10 CFR 50.59(c)(2), the change can be made without obtaining a license amendment pursuant to 10 CFR 50.90.

SECTION 3

CORE RELOAD EVALUATION

EC 291256, REVISION 0

ST. LUCIE UNIT 2 CYCLE 24 RELOAD

SUMMARY

The primary design change to the core for Cycle 24 is the replacement of 117 irradiated Westinghouse fuel assemblies with 88 fresh Region EE Framatome (formerly AREVA) fuel assemblies and 29 Regions AA and BB burned Westinghouse fuel assemblies. The Cycle 24 core will thus contain 88 fresh assemblies, and 129 previously burned fuel assemblies (100 Region DD once burned Framatome assemblies, 16 Region BB twice burned Westinghouse assemblies discharged from Cycle 22, and 13 Region AA twice burned Westinghouse assemblies discharged from Cycle 21).

The Reload EC 291256 describes a change made to the Biasi Critical Heat Flux (CHF) correlation Departure from Nucleate Boiling Ratio (DNBR) limit used in the post-scram Main Steam Line Break (MSLB) analysis. In this change, the design limit DNBR for post-trip MSLB was changed to a more conservative value based on the new available data and analysis. The calculated analysis minimum DNBR value continues to conservatively meet the limit. This change to the correlation limit screened in for 50.59 evaluation as a change to an element of analysis methodology used in performing safety analyses described in the UFSAR.

The only change to consider in the 10 CFR 50.59 Evaluation was the change in Biasi CHF DNBR limit. Since the change in Biasi CHF DNBR limit involves only a change to the method of evaluation, the 10 CFR 50.59 evaluation should only address criterion viii of 10 CFR 50.59(c)(2), and criteria 10 CFR 50.59(c)(2)(i—vii) are not applicable, consistent with NEI 96-07, Revision 1, Section 4.3.8 guidance.

For purposes of evaluations under criterion viii, the following changes are considered a departure from a method of evaluation described in the UFSAR:

1. Changes to any element of analysis methodology that yield results that are non-conservative or not essentially the same as the results from the analyses of record, or
2. Use of new or different methods of evaluation that are not approved by NRC for the intended application.

The change in Biasi CHF DNBR limit in the MSLB analysis does not constitute a new or different method of evaluation, so criterion #2 from above is not applicable.

For criterion #1, the Biasi CHF DNBR limit is changed to a more conservative value such that the margin from the analytical value to the DNBR limit has been reduced. The change in limit thus constitutes a conservative result relative to the previous limit. Therefore, the change is not a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

No activity requiring prior NRC approval per 10 CFR 50.59 was identified and no Technical Specification change is involved. The change in the Biasi Critical Heat Flux correlation DNBR limit used in the Main Steam Line Break analysis can be implemented without prior approval by the NRC.