September 16, 2020

L-2020-124
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 26 of the Updated Final Safety Analysis Report (UFSAR) (October 3, 2018 through March 20, 2020) dated September 2020. Amendment 26 of the UFSAR is submitted under a separate cover letter.

Please contact me at 772-467-7435 with any questions regarding this submittal.

Sincerely,

Wyatt Godes
Licensing Manager
St. Lucie Plant

WG/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
OCTOBER 3, 2018 THROUGH MARCH 20, 2020

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 UFSAR;
ii) changes in procedures as described in the UFSAR; and
iii) tests and experiments not described in the UFSAR

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 the period of October 3, 2018 through March 20, 2020.

This report is divided into three (3) sections:

1. Summaries of changes to the facility as described in the UFSAR performed by a permanent modification are summarized.
2. Summaries of changes to the facility or procedures as described in the UFSAR, and for tests and experiments not described in the UFSAR, which are not performed by a permanent modification.
3. A summary of any fuel reload 10 CFR 50.59 evaluation.

Sections 1, 2 and 3 summarize specific 10 CFR 50.59 evaluations for the specific changes. Each of these 10 CFR 50.59 evaluations concluded that the change did not require a change to the plant technical specifications, and prior NRC approval was not required.

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SECTION 1

PERMANENT MODIFICATIONS

EC 292875
2A1 6.9 kV SWITCHGEAR AND CIRCUIT BREAKER REPLACEMENT

SUMMARY

EC 292875 replaced the existing 2A1 6.9 kV Switchgear and circuit breakers with a new switchgear with circuit breakers using vacuum technology. In addition to this change the new switchgear replaced electromechanical relays used to provide motor protection with multifunction digital relays. The 6.9 kV circuit breaker protection is described in the UFSAR as to the types of protection provided. The UFSAR (Section 8.3.1.1.1.a) states the bus is protected with differential relays and the individual circuit breakers are protected for overcurrent, short circuit and fault detection. These protective features were provided by multiple electromechanical devices. The modification maintains the types of protection Indicated in the UFSAR; however, these functions were combined into a single relay for each circuit breaker, except for the bus differential protection, which is a separate digital relay. The results of the 10 CFR 50.59 Screening determined that the use of the digital relays is adverse and requires an evaluation under 10 CFR 50.59.

This permanent modification will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR. The replacement relays will perform the same function as those described in the UFSAR. These functions are buss differential protection, overcurrent and short circuit current protection, undervoltage protection, and backup breaker failure detection. These relays are relied upon for pump motor protection. Loss of one or more reactor coolant pump or one feedwater pump is identified as a potential initiator of a Partial Loss of Normal Feedwater Flow transient or a Partial Loss of Forced Reactor Coolant Flow. These events are considered moderate frequency events. The replacement relays are highly reliable and are of equal or higher reliability than the non-nuclear safety (NNS), non-seismic, electromechanical relays they replaced. The replacement digital relays have the same classification as the existing relays. The proposed change reduces the number of protective relays from 37 electromechanical relays for the entire switchgear to 8 multi-functional digital relays. Consequently, the proposed change provides the same level of protection and reduces the potential for relay failure that could impact feedwater pump or reactor coolant pump operation.

This permanent modification 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. The change will continue to protect the switchgear and subsequent motors powered by the switchgear (i.e., Reactor Coolant Pump motors and Feedwater Pump motor) against electrical transients. However, the failure of a feedwater pump leading to a Partial Loss of Normal Feedwater Flow transient and the loss of the reactor coolant pumps leading to a Partial Loss of Forced Reactor Coolant Flow transient are already events analyzed in the UFSAR, a failure of the new digital relay system would not result in any new adverse effects that have not been previously analyzed.

This permanent modification does not result in more than a minimal increase in the radiological consequences of an accident or malfunction previously evaluated in the UFSAR. The 6. 9 kV electrical distribution system serves no safety function. The loss of normal feedwater flow, the loss of condenser vacuum and the partial loss of forced reactor coolant flow are analyzed events. Accident analysis shows that failure of the 6.9 kV electrical distribution system will not compromise any safety related systems or prevent safe shutdown. Radiological consequences are not impacted by the proposed change.

This permanent modification does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR. The proposed change affects the 2A 1 6.9 kV Switchgear circuit breakers and protection relays and the potential for a partial loss of normal feedwater flow transient or a partial loss of reactor coolant flow transient due to the loss of one of the motors or a complete loss of the 2A 1 6.9 kV Switchgear. These transients have already been analyzed in the UFSAR. Operation or potential failure of the replaced NNS components does not affect the analyzed sequence of events or the trip signals associated with accident events. The proposed change has no effect on the transient analysis. The Failure Modes and Effects Analysis did not identify any new types of system-level failure modes that could cause a different type of accident than previously evaluated in the UFSAR. The proposed change does not impact any other SSCs important to safety. Therefore, the proposed change will not create the possibility of an accident of a different type than previously evaluated in the UFSAR.

This permanent modification does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR. While a failure of the replaced NNS components could result in a complete loss of the 2A1 6.9 kV Switchgear, electrical system protection coordination has been performed to ensure the upstream protection relays limit the effects of an electrical fault to a single train. Therefore, the change would only affect the 2A1 switchgear or downstream loads and result in the same evaluated transients that are discussed above. The Failure Modes and Effects Analysis did not identify any new types of system-level failures (that are as likely to occur as those failures previously considered in the UFSAR) that would result in effects not bounded by the results previously considered in the UFSAR. Consequently, the proposed change does not introduce the possibility for a malfunction of an SSC important to safety with a different result that previously evaluated in the UFSAR.

This permanent modification does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. The change affects the non-safety related 6.9 kV electrical distribution system protection relays and their potential for initiating a loss of normal feedwater or partial loss of forced reactor coolant flow, which are anticipated operational occurrences, The proposed change does not impact the analyzed transients and does not result in any impact on the fuel cladding,

RCS boundary, or containment integrity. Therefore, the proposed activity has no effect on the design basis limits for these fission product barriers.

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 291159
ROD CONTROL UPGRADE

SUMMARY

EC 291159 replaced the existing reactor Control Element Assembly (CEA) Control System with a Westinghouse Advanced Rod Control Hybrid (ARCH) Digital Control System including Ovation Control Logic.

The accidents previously evaluated in the UFSAR that could potentially be affected by this activity are as follows:

- UFSAR Section 15.1.3 “Excessive Increase in Secondary Steam Flow (Excess Load)”
- UFSAR Section 15.2.2 “Turbine Trip”
- UFSAR Section 15.4.1 “Uncontrolled Control Element Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition”
- UFSAR Section 15.4.2 “Uncontrolled Control Element Assembly Bank Withdrawal at Power”
- UFSAR Section 15.4.3 “Control Element Assembly Misoperation”

The malfunctions of SSCs important to safety previously evaluated in the UFSAR that could potentially be affected by this activity are as follows:

1. Malfunctions resulting in loss or partial loss of RSPT based Rod Position Indication data used by the Operator for compliance with CEA alignment restrictions.
2. Malfunctions resulting in loss or partial loss of RSPT based Rod Position Indication alarms used to alert the Operator to an abnormal CEA alignment.
3. Malfunctions resulting in loss or partial loss of RSPT based interlocks and CEA Motion Inhibit signals.
4. Malfunctions resulting in loss or partial loss of Step Count based Rod Position Indication data used by the Operator for compliance with CEA alignment restrictions.
5. Malfunctions resulting in loss or partial loss of Step Count based Rod Position Indication alarms used to alert the Operator to an abnormal CEA alignment.
6. Malfunctions resulting in loss or partial loss of Step Count based interlocks and Sequential Permissive signals.
7. Malfunctions resulting in loss or partial loss of Core Mimic Rod Position Indication data used by the Operator for assessment of Dropped Rod events.
8. Malfunctions resulting in loss of CEA position control capability needed to maintain normal operating conditions (i.e. RCS temperature) in response to reactivity changes.
9. Malfunctions resulting in loss of CEA position control capability needed for compliance with CEA alignment restrictions.
10. Malfunctions resulting in spurious rod motion.
11. Malfunctions resulting in a dropped rod.
12. Malfunctions resulting in a Turbine Overspeed condition.

13. Malfunctions resulting in a Turbine Trip.
14. Malfunctions resulting in spurious opening of Turbine Governor or Throttle valves.

Qualitative Assessments have been performed for changes associated with the Rod Control, Rod Position Indication and Turbine Control Systems. With the failure likelihood introduced by the modified SSCs being sufficiently low, there is not more than a minimal increase in the frequency of occurrence of a malfunction of an SSC important to safety or of an accident previously evaluated in the UFSAR.

The accidents and SSC malfunctions previously evaluated in the UFSAR that could potentially be affected by this activity either do not have any resulting radiological consequences or those consequences are bounded by other events which are not adversely affected by this change involving the Rod Control and Turbine Control systems. Therefore, this activity does not result in more than a minimal increase in the radiological consequences of a malfunction of an SSC important to safety or of an accident previously evaluated in the UFSAR.

A qualitative assessment was prepared for each of the five major portions of this overall digital system upgrade (i.e. reed switch position transmitter (RSPT) based RPI, Step Count based RPI, Core Mimic, Rod Control System and Turbine Control System). Each of the five qualitative assessments concluded that the failure likelihood introduced by the changes made to the Rod Control and Turbine Control Systems is sufficiently low. As such, the activity does not introduce any failures that are as likely to happen as those in the UFSAR that can initiate an accident of a different type. Therefore, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

A detailed evaluation of the failure modes and effects of this design change is contained in the qualitative assessment discussed above. The overall conclusion is that there is no credible failure that causes an adverse effect to the Rod Control or Turbine Control Systems. With the failure likelihood introduced by the modified SSCs being sufficiently low, the activity does not introduce any failures that are as likely to happen as those in the UFSAR that can initiate a malfunction of an SSC important to safety. Therefore, the activity does not create a possibility for a malfunction of an SSC important to safety with a different result from any previously evaluated in the UFSAR.

There are no fission product barrier design basis limits that are associated with or affected by this activity.

There are no methods of evaluation described in the UFSAR that are associated with or affected by this activity.

Regarding Technical Specifications, the following sections of the COLR are associated with this activity:

- Section 2.2: Full Length CEA Position - Misalignment > 15 Inches (TS 3.1.3.1)
- Section 2.3: Regulating CEA Insertion Limits (TS 3.1.3.6)
- Figure 3.1-2: CEA Insertion Limits vs. THERMAL POWER

The replacement Rod Control System and Rod Position Indication System will comply with all COLR requirements. There is no adverse impact on the COLR as a result of this activity.

The following sections of the Technical Specifications are associated with this activity:

- TS 3/4.1.3: Moveable Control Assemblies – Full Length CEA Position
- TS 3.1.3.1: CEA Block Circuit and Full Length (shutdown and regulating) CEAs
- TS 3.1.3.2: CEA Reed Switch and Pulse Counting Position Indicator Channels
- TS 3.1.3.4: CEA Drop Time
- TS 3.1.3.5: Shutdown CEA Insertion Limit
- TS 3.1.3.6: Regulating CEA Insertion Limit

The replacement Rod Control System and Rod Position Indication System will comply with all Technical Specification requirements. There is no adverse impact on the Technical Specifications as a result of this activity.

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

50.59 EVALUATIONS

EC 292636
FUEL TRANSITION NON-LOCA EVENTS UFSAR CHANGES

SUMMARY

EC 292636 documents the results of the Framatome UFSAR Chapter 15 reanalyses for the events that were not affected by the Westinghouse to Framatome fuel transition, and it revises the affected UFSAR sections to reflect these newly completed analyses. The Cycle 25 reload design is consistent with these new analyses. The EC changes addressed in this 50.59 Evaluation, for implementation in Cycle 25, include:

1. UFSAR changes associated with revised accident analyses.
2. New and revised operator actions assumed in the revised accident reanalyses.
3. Engineering evaluation changes associated with items 1 and 2 above (PSL-ENG-SEMS-12-006).
4. Procedure changes associated with items 2 and 3 above.
5. New Framatome methodology for accident reanalysis, instead of the previous Westinghouse methodology.

This activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR. There are no changes to plant physical configuration, and no SSCs along with their function are adversely affected, so the frequency of occurrence of previously evaluated accidents has not increased. Thus, the analyzed events remain with the same accident frequency categories and do not change the accident frequency within their current categories.

This activity does not result in more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. The SSC malfunctions assumed in the design basis accidents described in the UFSAR have not been altered or affected by the new or revised operator actions used in the safety analysis. This activity does not alter, modify or introduce new plant equipment.

This activity does not result in more than a minimal increase in the radiological consequences of an accident previously evaluated in the UFSAR. For all the analyzed events, the accident analyses acceptance criteria continue to be met, and radiological consequences are not affected.

This activity does not result in more than a minimal increase in the radiological consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR. The SSC malfunctions assumed in the design basis accidents described in the UFSAR have not been altered or affected by new or revised operator actions used in the safety analysis.

This activity does not alter, modify or introduce new plant equipment. The activity is only concerned with the accident reanalyses using Framatome NRC approved methodology and new or revised operator actions used in the affected safety analysis. As a result, the malfunctions of SSCs assumed in the accident analyses are not impacted, so no new malfunctions are created. Therefore, the proposed activity does not create the possibility for either an accident of a different type or for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

This activity does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. In the various accidents that were analyzed, the main parameters that were affected include: RCS pressure (to avoid exceeding RCS design pressure boundary limit), main steam system (MSS) pressure (to avoid exceeding MSS design pressure boundary limit), RCS subcooling margin (to maintain integrity of fuel), pressurizer level (to avoid overfill), RCS water inventory (to maintain natural circulation), and SG level (to maintain heat sink). These parameters have established values and acceptance criteria limits provided in the UFSAR for the various accidents to ensure that the design basis limits for the fission product barriers remain intact. The EC 292636 design evaluation demonstrates that none of these design basis limits for fission product barriers have been exceeded with the new operator actions used in the safety analysis.

This activity does 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. The use of new or revised operator actions does not constitute a departure from the method of evaluation described in the UFSAR. The new Framatome methodology used in the safety analyses has been approved by the US NRC for St. Lucie application as part of the fuel transition License Amendment.

No Technical Specification change is required.

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 293175

TURBINE VALVE TESTING INTERVAL CHANGE TO NINE MONTHS

SUMMARY

Turbine valve testing ensures that these valves will reliably close when required during a turbine overspeed event. EC 293175 changes the St Lucie Unit 2 turbine valve testing interval from six months to nine months.

This activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR. Per Siemens Technical Report CT-27455 Rev 1, the current total probability of an external missile for the unit at 100,000 hours of inspection interval is 1.88E-6 with a 6-month Turbine Valve Testing Interval. Using the probability of overspeed per year listed in CT-27455 Rev. 1 with a polynomial curve fit, the calculated total probability of an overspeed with a 9-month testing interval is 3.45E-6. According to NEI 96-07 Rev. 1 Section 4.3.1, a licensee shall remain below plant specific criteria. The NRC set limit for probability of an external missile is 1.0E-5 per year or 11.42E-5 per 100,000 hours. A probability of 3.45E-6 for an external missile is less than NRC required limit and is therefore acceptable.

The total probability of an overspeed per year increases from 1.88x10-6 for a 6-month turbine valve testing interval to 3.45E-6 for a 9-month testing interval. The increase is by a factor of 1.84. Per NEI 96-07 Rev 1, a change is considered adverse if the change in likelihood of occurrence of a malfunction increases by more than a factor of two. Since the total probability of an external missile increases by a factor of 1.84 times by changing the turbine valve testing interval to 9 months, this change is not considered more than minimal; therefore, the change is acceptable.

Failure of the turbine stop and control valves to close and prevent a turbine overspeed event are the only malfunctions that could credibly occur due to this activity. These turbine malfunctions do not involve a radiological consequence nor are any radiological consequences postulated as a result of a turbine missile event. Because the probability of occurrence of a turbine missile accident remains within plant specific NRC criteria for this activity, the potential for unacceptable damage is precluded and no increase in radiological consequences are postulated.

The turbine missile is the only accident previously evaluated in the UFSAR that is credibly affected due to this activity. No new failure modes are introduced. Failure of the turbine stop and control valves to close are not an initiator of any accidents other than a turbine missile accident. As such, this activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

The change in turbine valve test frequency from 6-month intervals to 9-month intervals does not introduce the possibility for a malfunction of an SSC with a different result because the activity does not introduce any new failure modes.

Fission barrier integrity is not adversely impacted by a postulated turbine missile accident. Therefore, this change does not affect a design basis limit for a fission product barrier.

This activity relies on the methodology developed by Siemens Energy Inc. and approved by NRC as described in Section 3.5.1.3.2.2 of the Unit 2 UFSAR. Therefore, this activity does not constitute a departure from a method of evaluation described in the UFSAR.

No Technical Specification change is required.

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

FUEL RELOAD EVALUATION

EC 293944

ST. LUCIE UNIT 2 CYCLE 25 RELOAD

SUMMARY

The St. Lucie Unit 2 Cycle 25 Core Reload was mainly the reconfiguration of the reactor core to replace a portion of the fuel assemblies in the reactor to enable Unit 2 to operate at full power for Cycle 25 to meet the energy requirements of the approved operating schedule. The primary design change to the core for Cycle 25 was the replacement of 81 irradiated fuel assemblies with 81 fresh Region FF Framatome (formerly AREVA) fuel assemblies. The Cycle 25 core contains 81 fresh assemblies, and 136 previously burned fuel assemblies (88 Region EE once-burned assemblies, and 48 Region DD twice-burned assemblies). All Cycle 25 assemblies are Framatome fuel previously approved by the NRC in License Amendment 182.

Other cycle specific changes (e.g., UFSAR and procedure changes associated with the removal of the mini-dual CEAs, which was approved by the NRC in License Amendment 198) did not require a 10 CFR 50.59 Evaluation. The discussions within this EC, along with the 10 CFR 50.59 checklist, justify that the design and operation of the Cycle 25 reload core meets the 10 CFR 50.59 (c)(2) criteria.

The core reload activities were implemented with no additional changes to the St. Lucie Unit 2 Technical Specifications other than those approved by the NRC in License Amendments 182 and 198.

It was concluded that this reload complies with all the applicable Technical Specifications/design basis limits and per 10CFR50.59 (c)(2) criteria, thus, no prior NRC approval was needed for the implementation of this reload.