FPL

April 19, 2016

L-2016-088
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 evaluation of each, which were made on Unit 2 during the period of Amendment No. 23 (April 24, 2014 through October 26, 2015) dated April 2016 and subsequent minor updates 22A and 22B. This submittal correlates with the information included in Amendment 23 of the Updated Final Safety Analysis Report to be submitted under separate cover.

Please contact us if there any questions on this information.

Sincerely,

Michael J. Snyder
Licensing Manager
St. Lucie Plant

MJS/lrb

Enclosure

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

IE47
NRR

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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 23 addressing the period of April 24, 2014 through October 26, 2015 and subsequent minor updates 22A and 22B.

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 Fuel 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

PLANT CHANGE / MODIFICATIONS

EC 283720, REVISION 5

2B2 SAFETY INJECTION TANK (SIT) DISCHARGE PIPING REPAIRS AND SUPPORT MODIFICATION

SUMMARY:

On 4/11/2015 it was identified in Condition Report (AR2039830) that a Unit 2 through-wall flaw in 2B2 safety injection tank (SIT) discharge header piping I-12"-SI-459 was discovered. The flaw was identified during walkdowns to determine the source of SIT 2B2 leakage. The flaw was at support SI-4203-44 in the Class 2 piping beneath the 2B2 SIT as depicted on Flow Diagram 2998-G-078, sheet 132. EC 283720 was implemented to address removal of I-12"-SI-459 in Mode 3 and restoration of the piping and supports. The 2B2 SIT replacement piping was identified to have a thicker wall than the specified manufacturer's tolerance. As a result, the inside diameter of the piping was slightly less than the original piping. The slightly reduced inside diameter of the replacement piping adversely impacts the piping friction coefficient "K" of the SIT discharge piping.

Reduction in the pipe inside diameter will not increase the frequency of occurrence of an accident previously evaluated in the UFSAR, nor will it will not increase the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. As documented in AR 02040616, the replacement piping has a thicker wall in excess of the manufacturer's tolerance for 12" schedule 40S piping (appears to meet schedule 40 thickness tolerance). As a result, the inside diameter of the piping will be slightly less than the original piping. Replacement of a section of I-12"-SI-459 with piping having a slightly reduced inside diameter adversely impacts the piping friction coefficient "K" of the SIT discharge piping. However, a review of Design Basis Document DBD-HPSI-2 and the calculation for resistance in the piping NSSS-026 shows that the entire piping system from the SIT tank to the RCS was considered to be Schedule 160 (area =0.5592 sq ft), with a resultant K=8.78 (maximum). The St. Lucie Unit 2 Cycle 21 Ground rules input is K=8.78 (maximum). Therefore, reduction in the pipe inside diameter is bounded by the accident analysis input parameters. As a result, the proposed activity will not result in an increase in the radiological consequences of an accident previously evaluated in the UFSAR.

Reduction in the pipe inside diameter is bounded by the accident analysis input parameters for the resistance considered in this piping. As a result, the proposed activity will not result in an increase in the radiological consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR. There are no new components added that could create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

Reduction in the pipe inside diameter is bounded by the accident analysis input parameters for the resistance considered in this piping. There are no new components added that could create a possibility for a malfunction of an SSC

important to safety with a different result than any previously evaluated in UFSAR. Reduction in the pipe inside diameter is bounded by the accident analysis input parameters for the resistance considered in this piping. Therefore, the proposed activity does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. The modification involves a pipe replacement and support rework. In addition, the proposed activity 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 changes in the method of evaluation as a result of this modification.

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 281771, REVISION 7

REPLACEMENT OF REACTOR COOLANT PUMP (RCP) 2A2 ROTATING ASSEMBLY

SUMMARY:

To allow for future maintenance of the reactor coolant pump the RCP 2A2 whip (cable) restraints were permanently removed. In addition to removing an interference that impedes the disassembly of the pump and in-service inspections (ISI) of the RCP inlet nozzle dissimilar weld it will also reduce the radiological dose required to reinstall the 4-inch cable.

The permanent removal of the upper two 4-inch reactor coolant pump RCP 2A2 whip (cable) restraints meets the acceptance criteria found in Generic Letter 87-11 in that as documented in NRC letter dated March 5, 1993, the NRC staff concluded that since the St. Lucie Units are bounded by the CEOG analyses and the leakage detection systems are capable of detecting the specified leakage rate, the dynamic effects associated with postulated pipe breaks in the primary coolant system piping can be excluded from the licensing and design bases of the St. Lucie Units. The EPU leak-before-break evaluation is based on evaluation CEN-367-A. The primary loop piping normal operating, SSE and pressure loads due to the EPU conditions were used in the EPU evaluation. The results of the evaluation demonstrated that leak-before-break recommended margins used for the primary loop piping continue to be satisfied for the EPU conditions. Calculation CN-MRCDA-09068 provides an analysis of the RCP and surrounding components to demonstrate the RCP HELB cable restraints for 2A1, 2A2, 2B1 and 2B2 can be permanently removed. In addition, Calculation CN-MRCDA-09-36 documetns that sufficient leak-before-break margin on crack stability is maintained on the main coolant loop hot leg and cold leg pipes under EPU conditions. The NRC accepted leak-before-break for EPU conditions on St. Lucie Unit 2 per letter dated September 24, 2012. The permanent removal of the upper RCP 2A2 whip restraints meets the acceptance criteria of BTP 2-1, as contained in SRP Section 3.6.2, in that absent the whip restraints the primary coolant system piping continues to meet the applicable ASME Code design requirements.

As described in UFSAR Section 3.8.3.1.5, the upper RCP whip (cable) restraints are provided to restrain the reactor coolant pumps in the unlikely event of a break in the Reactor Coolant System piping. Accordingly, the RCP 2A2 upper whip restraints are not related to accident precursors or accident initiators and, therefore, their permanent removal will not result in any increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The removal of the RCP cable restraints does not introduce the possibility of a change in the likelihood of a malfunction because the wire rope restraints are provided to restrain the reactor coolant pumps in the unlikely event of a break of the

Reactor Coolant System piping (UFSAR Section 3.8.3.1.5). Additionally, with the implementation of leak-before-break (UFSAR Section 3.6) the dynamic effects associated with the postulated pipe breaks in the primary coolant system are no longer part of the licensing and design bases of the St. Lucie Units. Therefore, the restraints are no longer required and accordingly essential systems, components, and structures are still considered to be adequately protected against the effects of postulated piping failures, which no longer include guillotine and slot breaks.

The removal of the RCP cable restraints does not introduce the possibility of a change in the consequences of an accident because the dynamic effects of guillotine and slot breaks in the RCS hot and cold leg piping have been eliminated due to leak-before-berak (LBB) technology and the cable restraints are no longer required (UFSAR Section 3.6).

With the elimination of the dynamic effects associated with a hot or cold leg break essential systems, components and structures are still adequately protected and, therefore, the radiological consequences of an accident previously evaluated has not increased.

The removal of the RCP cable restraints does not introduce the possibility of a change in the consequences of a malfunction because the cable restraints are no longer required to mitigate dynamic effects of guillotine and slot breaks in the RCS hot and cold leg piping due to the adoption, and NRC approval, of LBB technology (UFSAR Section 3.6).

The removal of the RCP whip restraints does not introduce the possibility of a new accident because the whip restraints are not an initiator of any accident and no new failure modes are introduced.

The removal of the RCP whip restraints does not introduce the possibility for a malfunction of an SSC with a different result because the activity does not introduce a failure result.

The RCP cable restraints were originally installed to mitigate the consequences of the dynamic effects associated with a hot or cold leg break. However, with the adoption of, and the approval by the NRC, on the leak before break methodology (UFSAR Section 3.6) the dynamic effect of guillotine and slot breaks in RCS hot and cold leg piping are no longer considered a plant design basis. With the elimination of these dynamic effects, along with the whip restraints, the essential SSCs are still considered to be adequately protected under LBB.

The original criteria for the whip restraints were to restrain the reactor coolant pumps in the unlikely event of a break of the Reactor Coolant System piping. This criterion is no longer applicable with Leak-Before-Break and the whip restraints can be eliminated. Hence, a design basis limit for a fission product barrier will not be exceeded or altered.

In summary, the removal of the RCP whip restraints will not result in a design basis limit for a fission product barrier being exceeded or altered.

The removal of the RCP whip restraints involves a method of evaluation as defined in UFSAR Section 3.6. Although the original design and licensing bases included the restraints, the approval of Leak-Before-Break by the NRC allowed the dynamic effects associated with a hot or cold leg break to be eliminated from the plant design basis. The technical justification was provided by CEN-367-A, "Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems," which was endorsed by the NRC in 1990 and later shown to bound the St. Lucie Units; LBB was approved by the NRC in 1993. The CEN-367-A evaluation still justifies the removal of the restraints. Calculation CN-MRCDA-09-68 provides an analysis of the RCP and surrounding components to demonstrate the RCP HELB cable restraints for pumps 2A1, 2A2, 2B1 and 2B2 can be permanently removed. In addition, Calculation CN-MRCDA-09-36 documents that sufficient leak-before-break margin on crack stability is maintained on the main coolant loop hot and cold legs under EPU conditions. The NRC accepted leak-before-break for EPU conditions on St. Lucie Unit 2 per letter dated September 24, 2012. Therefore a departure from the method of evaluation is not required.

There are no Technical Specifications that address the restraints. However, as discussed in the Safety Evaluation by the NRC on Leak-Before-Break (LBB) Technology, the acceptance of LBB is based on a leakage detection system consistent with Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." Technical Specification Section 3/4.4.4.6.1 addresses the RCS leakage detection system and no changes to this section or any other section is required as a result of the removal of the cable restraints.

The removal of the pipe whip restraints was previously approved by the NRC when the Leak-Before-Break Technology was approved and as documented in UFSAR Section 3.6, the mechanical/structural loads associated with the dynamic effects of guillotine and slot breaks in RCS hot and cold leg piping are no longer considered a plant design basis. 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 285101, REVISION 0

INCREASED STEAM GENERATOR BLOWDOWN FLOW RATE (UNIT 2)

SUMMARY:

The EC 285101 associated with this 50.59 Evaluation has been prepared to increase the current steam generator blowdown (SGBD) system flow rate values in the accident analysis from 50 gpm per SG to a maximum value of 120 gpm per SG, and make the necessary UFSAR changes as a result. Also, an UFSAR change is processed to clarify timing for opening the hot leg injection valves in the accident analysis. The activity has been considered to be ADVERSE per 50.59 screening, since it affects the maximum SGBD flow rate assumed in the accident analyses for the loss of feedwater (LOMF) event, feedwater line break (FLB) event, and Station Blackout (SBO) event, and operator action time requirements have been assumed for the LOF event. This evaluation is applicable to the UFSAR changes, DBD changes, procedure changes, Operator Action Times evaluation PSL-ENG-SEMS-12-006 changes, and new safety analyses described in Section 2.3 of the EC, which are the new design basis.

The previously analyzed accidents in the UFSAR for LOMF/SBO (Chapter 15) and FLB (Chapters 10 and 15) were reanalyzed and the results of the reanalysis show compliance with the acceptance criteria for these events, thus, there is no change in the frequency of occurrence category as prior to the changes for the maximum SGBD system flow rates in all events, with the newly assumed operator action times in the LOMF event. Therefore, 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.

The accident scenarios that were considered are those that were reanalyzed, since the changes to the maximum SGBD system flow rates in the LOMF, FLB and SBO events, and operator action times assumed in the LOMF event may only have an effect on those accidents. In these analyses, several conditions were postulated, including: loss of offsite power (LOOP), loss of 125v AC bus, pipe break disabling the turbine driven AFW pump, and failure of a motor-operated AFW pump in the Chapter 10 LOMF event; loss of a turbine driven AFW pump for the Chapter 15 LOMF and FLB events; and LOOP coincident with turbine trip and failure of all emergency diesel generators for the SBO event. However, none of the EC changes affect the previously assumed malfunctions in these analyses or the operation of any SSC important to safety relied upon to mitigate the consequences of these accidents. Therefore, the proposed activity does 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 previously analyzed accidents in the UFSAR affected by the analysis changes are the LOMF/SBO (Chapter 15) and FLB (Chapter 15). As the results of these re-analyses show that the acceptance criteria have not been violated, there is no impact on the

current analyses of record DNBR or FCM SAFDLs due to the parameter changes. Thus, the radiological consequences for these events remain unchanged. Therefore, the proposed activity does not result in more than a minimal increase in the radiological consequences of an accident previously evaluated in the UFSAR.

The previously analyzed accidents in the UFSAR affected by the analysis changes are the LOMF/SBO (Chapter 15) and FLB (Chapter 15). For these reanalyzed events, the changes for the maximum SGBD system flow rates and operator action times assumed in the analyses do not affect the operation of equipment or components considered in these analyses. The Offsite Dose Calculation Manual (ODCM) establishes radioactive release rates to comply with radioactive release limits provided in 10 CFR 20 and 10 CFR 50. The ODCM program methodology section establishes the criteria and justification for SGBD radiation monitor setpoints for the unit with a maximum blowdown flow rate of 125 gpm per SG, which bounds the maximum proposed flow rate in this evaluation. Thus, the radiological consequences for these events remain unchanged. Therefore, the proposed 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 changes in this EC including the maximum SGBD system flow rates in the analyses and operator action times were reanalyzed for the accidents in the UFSAR as described above. Based on the analyses results, there is no negative effect on any plant systems, including the SGBD system. As a result, the activity does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR.

The changes in this EC, including the maximum SGBD system flow rates in the analyses, and operator action times were reanalyzed for the accidents in the UFSAR as described above. These parameters do not have an effect on any plant systems, including the SGBD system. As a result, they do not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

The changes in this EC including the maximum SGBD system flow rates in the analyses, and operator action times were reanalyzed for the accidents in the UFSAR, as described above. The results of these re-analyses show no impact on DNBR, FCM SAFDLs, reactor coolant system or containment design basis limits. Therefore, there is no impact on the fuel cladding, RCS pressure boundary or containment integrity. In conclusion, the proposed activity does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

The design bases methods and evaluation methodology used for the reanalysis of the LOMF/SBO (Chapter 15) and FLB (Chapter 15) events as a result of the increase in the maximum SGBD system flow rates and operator action times are the same as those used for the analysis of record. The re-analyses performed as a result of the changes in this EC were performed with the same method of evaluation used for previous safety analyses. Therefore, the proposed activity does not involve replacing or revising an

UFSAR described evaluation methodology used in establishing the design basis or used in the safety analysis.

Technical specifications review concludes that no Technical Specifications are affected by 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 3

CORE RELOAD EVALUATION

EC 283959, REVISION 0
ST. LUCIE UNIT 2 CYCLE 22 RELOAD

SUMMARY

St. Lucie Unit 2 Cycle 22 Core Reload did not require a 10 CFR 50.59 Evaluation. The discussions within this EC, along with the 10 CFR 50.59 Applicability/Screening which were performed, justify that the design and operation of the Cycle 22 reload core will meet the 10 CFR 50.59 (c)(2) criteria. The core reload activities can be implemented with no changes to the St. Lucie Unit 2 Technical Specifications. The safety analyses results are within the current design basis, within the acceptance limits provided by the NRC regulatory criteria and within the criteria provided by 10 CFR 50.59. Therefore, prior NRC approval is not required for implementation of this EC.