FPL.

May 23, 2013

L-2013-175
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 50.59 evaluation of each which were made on Unit 2 during the period of May 8, 2011 through November 23, 2012. This submittal correlates with the information included in Amendment 21 of the Updated Final Safety Analysis Report to be submitted under separate cover.

Please contact us should there by any questions regarding this information.

Sincerely,

E Katzman

Eric S. Katzman
Licensing Manager
St. Lucie Plant

ESK/tlt

Attachment

IE47
NRR

ST. LUCIE UNIT 2
DOCKET NUMBER 50-389
CHANGES, TESTS AND EXPERIMENTS
MADE AS ALLOWED BY 10 CFR 50.59
FOR THE PERIOD OF
MAY 8, 2011 THROUGH NOVEMBER 23, 2012

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 the period of May 8, 2011 through November 23, 2012.

This report is divided into two (2) sections. First, changes to the facility as described in the Updated Final Safety Analysis Report (UFSAR) performed by a permanent modification either a Plant Change/Modification (PC/M) or an Engineering Change (EC). 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.

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

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SECTION 1

PERMANENT MODIFICATIONS

PLANT CHANGE/MODIFICATION EC246487

REVISION 1

<u>TCW HEAT EXCHANGER REPLACEMENT</u>

Summary:

EC246487 implemented activities that prepared the plant for operation at Extended Power Uprate (EPU) conditions, as described in St. Lucie Unit 2 EPU LAR (L-2011-021). The Turbine Cooling Water (TCW) System provides a heat sink for power cycle equipment and the purpose of EC246487 is to modify the TCW System to increase its cooling capacity under EPU conditions. Specifically, EC246487:

- Replaced the 2A and 2B turbine cooling water heat exchangers with larger units,
- Modified connected piping including relocation of instrument racks,
- Modified the heat exchanger pedestals,
- Installed two new flow indicators, and associated power supplies in each train of TCW,
- Replaced restriction orifices SO-21-2A and SO-21-2B to increase flow through the TCW heat exchangers.

The activities associated with implementation of EC246487 were reviewed against the UFSAR to identify SSC design functions that would be adversely affected by implementation of this modification, including impacts to structures, systems, and components (SSCs) identified in the EC246487 Design Change Checklist. Such assessments considered impacts to events and sequences important to safety analyses, including seismic analysis and accident analysis. For example, the modification increases the weight of TCW components, and changes the flow through from the ICW. The fire protection plan and UFSAR appendix 9.5A are not affected by this modification. The Turbine Cooling Water System is not discussed in UFSAR Chapter 6, Engineered Safety Features or Chapter 15, Accident Analysis. The TCW System has a primary interface with the non safety-related portion of the ICW System, including modifications to the Intake Cooling Water (ICW) System as a part of EC246487. Consequently, the design functions of both the TCW and the ICW systems were assessed for impacts resulting from implementation of EC246487.

The 10 CFR 50.59 Screening identified possible adverse effects to the functions of the Emergency Diesel Generator (EDG) system as discussed in the UFSAR, based on the increased EDG load from the ICW pump motors, which are included in the Emergency Diesel Generator (EDG) load and load sequencing calculation. Operating further out on the pump curve increases EDG loading and impacts EDG load sequencing because it decreases pump head and increases motor brake horsepower. The increase in brake horsepower will require an increase in electrical demand (current and voltage) from the 4.16kV buses. These impacts are considered adverse and therefore, this aspect of EC246487 activities was further investigated The proposed increase in ICW flow to the tube side of the new TCW heat exchangers results in an increase in the brake-horsepower for the ICW pumps, which are powered by the safety related 4.16kV buses. The increase in brake-horsepower will require an increase in electrical demand (current and voltage). The increase in electrical demand has two potential adverse consequences. First, the increased load from the ICW pumps could impact the Emergency Diesel Generator (EDG) load sequence during a design basis event. This impact results from higher ICW pump brake horsepower demand during normal operation that continues until the TCW is isolated on SIAS. By the time the ICW pumps are sequenced onto the 4.16kV bus (EDG Load Block 4; 9 seconds), the TCW isolation valves have not fully closed (70 seconds), creating the potential for an EDG malfunction.

Second, the increased electrical demand adversely impacts EDG fuel rate consumption and overall fuel capacity requirements, creating the potential for component malfunctions.
However, per Calculations PSL-2-FJE-90-0020-R10 and PSL-2FSE-03-010, analysis of the electrical system performance demonstrates that the increase in electrical demand is acceptable for proper equipment performance, does not compromise emergency diesel generator (EDG) capacity or load sequencing. (Subject to the Restrictions stated in Section V.) Therefore, the proposed activities will not result in more than a minimal increase the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

A review of the Unit 2 Technical Specifications (TS) for requirements relevant to the TCW System was performed. The TCW System heat exchangers do not perform a safety related function, are not required for safe shutdown and thus, are not described in the Unit 2 Technical Specifications. TCW System interfaces with the shutdown cooling heat exchanger and the ICW System. The proposed modifications do not affect any SIAS equipment actuation, including the isolation valves separating the non-nuclear safety portions of the ICW System from the safety related portions of the ICW System. TS 3.7.4 states that at least two independent ICW loops shall be OPERABLE in Modes 1, 2, 3 and 4. However, no changes to the TS are required to implement EC246487. Therefore, the proposed modifications to the TCW System do not adversely impact any TS or Bases-described SSCs or their ability to perform their described design functions and a change to the TS is not required.

The 10 CFR 50.59 Screening documents demonstrate that this activity does not require any changes to the Technical Specifications. The 10 CFR 50.59 Screening identified possible adverse effects to the functions of system(s) as discussed in the UFSAR; therefore, a 10 CFR 50.59 Evaluation was performed. The 10 CFR 50.59 Evaluation demonstrated that no prior NRC review and approval is required.

PLANT CHANGE/MODIFICATION EC246506

Permanent Removal of Seal Injection from Unit 2 RCPs

REVISION 18

Summary:

Modification EC246506 was performed on Reactor Coolant Pumps 2A1, 2A2, 2B1, and 2B2. This modification made the following changes:

1. Replaced the existing seal with a seal design with integral flanges.
2. Removed seal injection.
3. Modifies the middle and upper seal lines to add flexible hoses.
4. Modified the CBO line to add flexible hoses.
5. Modified the charging line supplying seal injection.
6. Increased the pressure in the CBO cavity.
7. Lowered the vapor seal line 1-5/8".
8. This modification addressed the abandonment strategy of seal injection.
9. This modification removed valves V2185 and V2598 from the Essential Equipment List and their associated cables from the Appendix 'R' Safe Shutdown Analysis.

The RCP seal is described in UFSAR consists of a seal cartridge which consists of four face type mechanical seals, three full pressure seals mounted in tandem and a fourth low pressure vapor seal designed to withstand system operating pressure when the pumps are not operating. A controlled bleed off flow through the seals is used to cool the seals and to equalize the pressure drop across each seal. The controlled bleed off flow is collected in the volume control tank of the chemical and volume control system. Leakage past the vapor seal is collected in the reactor cavity sump. Each seal is designed to accept the full operating pressure of the reactor coolant system. However, the first three seals of the cartridge assembly normally operate with a pressure differential equal to one-third of the operating pressure and with only a slight pressure differential across the vapor seal. The UFSAR also states that the seals are provided with a seal injection system. The existing seals have standard flanges on the end of approximately 7" long pipe stubs. The weight of the flanges on the end of these pipes, which are welded to the seal body, and the pump vibration create the potential for low stress high cycle fatigue. The modification of the seals eliminates this condition. The new seals have flange faces machined into the body of the seal eliminating the 7" long pipe stubs that cracked at the pipe stub to seal weld on two occasions. The change is structural. The design of the seal with respect to performing its hydraulic functions is unchanged. The description of the seal in the UFSAR is unchanged by this modification. The removal of seal injection from each RCP is not a change to the UFSAR with respect to using seal injection for seal cooling when component cooling water to the seal is lost. However, it is a change to eliminate seal injection with respect to using seal injection to prevent the introduction of debris into the seal during RCS fill and vent. This is considered a potentially adverse effect and required a 50.59 evaluation.

EC 246506 (PC/M 09106) removes the Seal Injection from Unit 2 Reactor Coolant Pumps 2A1, 2A2, 2B1, and 2B2.

Background:

During operation of St Lucie Unit I in 1977 and 1980, Component Cooing Water (CCW) to the

Reactor Coolant Pump (RCP) seal heat exchangers was lost, requiring the replacement of the RCP seals. As a result, an independent seal injection system was added on Units 1 & 2 to provide backup cooling to the RCP seals in case CCW was lost. Subsequent to the original design intent, the seal injection system is also used during filling of the RCS to prevent any debris in the RCS from entering the RCP seal. During the early 1990's, cracking was discovered in the RCP shafts caused by seal injection of cool water from the Chemical Volume & Control System (CVCS). As a result, seal injection was discontinued with exception of during filling of the RCS.

The PSL Unit 2 UFSAR states, "A reactor coolant pump (RCP) seal injection system is provided from the chemical and volume control system (CVCS). This seal injection system has capability to inject water into the RCP seals originally designed as a backup system for loss of Component Cooling Water (CCW). According to safety evaluation JPN-PLS-SENJ-93-001, RCP seal injection is used only during RCS fill and vent operations. Also, this evaluation disallows use of RCP seal injection for plant heatups and cooldowns, and off-normal and loss of CCW procedures. With the installation of N9000 seals designed and tested to cope with Station Blackout coping times, RCP seal injection cooling is no longer needed."

The evaluation JPN-PSL-SENJ-93-001 Revision 1, "Safety Evaluation for Deletion of RCP Seal Injection" also stated that "although seal injection is not required, it should continue to be used for the fill and vent operations".

The Nuclear Risk of retaining seal injection is considered greater than abandoning seal injection. Seal injection as currently used is as a precautionary measure to limit floating debris from entering the lower RCP seal cavity during RCS flood-up. As the benefits of RCP seal injection are minimal, the Operation Decision Making (ODM) team concluded that PSL seal injection can be eliminated.

Based upon above, the only intent of RCP seal injection as currently used is as a precautionary measure to limit floating debris from entering the lower RCP seal cavity during RCS flood-up from elevation below 33' to above 33'. The RCP seal is not expected to fail catastrophically as a result of expected debris entering the seal, the RCP seal injection is being abandoned. Therefore, this 10CFR 50.59 evaluation is prepared to evaluate the permanent removal of Seal Injection from all four pumps (i.e. RCPs 2A1, 2A2, 2B1 and 2B2).

There are no Technical Specifications that address the use of seal injection. Operability of the RCPs is addressed in Tech Spec Section 3/4.4, but seal injection is not required to be available to consider the RCP operable. Technical Specifications 3.4.6.2 discusses reactor coolant system operational leakage limits. There is no change to this specification because of the seal injection elimination. A License Amendment Request is not required. Seal injection that was originally designed to serve as a backup system for seal cooling has been evaluated as no longer needed as stated in UFSAR Section 9.3.4.2.1.2. The use of seal injection during fill and vent is of minimal value in preventing excess seal leakage.

The 10 CFR 50.59 Screening documents demonstrate that this activity does not require any changes to the Technical Specifications. The 10 CFR 50.59 Screening identified possible adverse effects to the functions of system(s) as discussed in the UFSAR; therefore, a 10 CFR 50.59 Evaluation was performed. The 10 CFR 50.59 Evaluation demonstrated that no prior NRC review and approval is required.

PLANT CHANGE/MODIFICATION EC249965

PSL2 Uprate – Electrical Bus Margin Improvement

REVISION 0

Summary:

Modification EC 249965 implemented activities that prepared the plant for operation at Extended Power Uprate (EPU) conditions, as described in St. Lucie Unit 2 EPU LAR (L-2011-021). The activities implement voltage margin improvements on the electrical busses of St. Lucie Nuclear Power Plant (PSL) Unit 2 which are decreased due to load added by Extended Power Uprate (EPU) modifications. The changes performed by this EC are as follows:

- Add a Safety Injection Actuation Signal (SIAS) trip of feedwater pumps 2A and 2B on Non-Nuclear Safety (NNS) 6.9 kV switchgear 2A1 and 2B1
- Add an SIAS trip of heater drain pumps 2A and 2B on 4.16 kV NNS switchgear 2A2 and 2B2
- Add an SIAS trip of Main Transformers 2A and 2B coolers on NNS 480 V switchgear 2A1 and 2B1
- Add an SIAS trip of the Generator Main Leads Fans 2A and 2B (Isophase bus cooling fans) on NNS 480 V switchgears 2A1 and 2B1
- Mount four electrical enclosures (B2532, B2533, B2534, and B2535) to install four relays (two Agastat EGPD isolation relays and two ABB MG-6 interposing relays) and associated terminal strips, fuse blocks and fuses
- Replace Current Limiting Reactors (CLRs) on 480V Switchgear 2A2, 2A5, 2B2, and 2B5
- Re-uses existing and adds new cables through conduits and cable trays
- Adds new Associated Circuits to the Appendix R Safe Shutdown Analysis (drawing 2998-B-048)

EC 249965 implemented voltage margin improvements on the electrical busses of PSL2 which are decreased due to load added by EPU modifications. The activities associated with implementation of EC249965 were reviewed against specific sections of the UFSAR to identify design functions that would be adversely affected. The UFSAR review also considered impacts to structures, systems and components (SSCs) identified in the EC249965 Design Change Checklist, under Battery Loading. The 125 VDC system and its components were evaluated to ensure they would be capable of performing their intended function at EPU conditions. The batteries will be impacted by the relays added for SIAS. The modification will increase the loads on the 125 VDC buses 2A and 2B. During an SIAS condition, there will be an additional load of 0.111 amps on each bus. Therefore, this aspect of the proposed modification has an impact on nuclear safety and required a 10CFR 50.59 evaluation.

SIAS is a safety function within ESFAS. The function of the ESFAS is to mitigate the consequences of an accident by initiating systems designed to provide core cooling, establish containment isolation, and maintain containment integrity once certain system operating parameters are exceeded. ESFAS utilizes output of instruments measuring these parameters to generate actuation signals. The SIAS is one such actuation signal which initiates when conditions are indicative of certain events, including a LOCA, to mitigate the consequences of these events. Parameters at these values will also initiate a reactor trip by the Reactor Protection System (RPS) using the same process transmitters. Using SIAS to trip the main feedwater pumps introduces the possibility for inadvertent spurious trip of a main feedwater pump. An increased frequency for a subsequent reactor trip would be an adverse affect during normal plant operation; thus, this aspect of the modification also required a 10CFR 50.59 evaluation.

The 10 CFR 50.59 Screening documents demonstrated that this activity does not require any changes to the Technical Specifications. The 10 CFR 50.59 Screening identified possible adverse effects to the functions of system(s) as discussed in the UFSAR; therefore, a 10 CFR 50.59 Evaluation was performed. The 10 CFR 50.59 Evaluation demonstrated that no prior NRC review and approval is required.

PLANT CHANGE/MODIFICATION EC249969

Heater Drain / MSR Digital Controls

REVISION 1

Summary:

The activities implemented by EC249969 support St. Lucie Unit 2 Extended Power Uprate (EPU) Licensing Amendment Request (LAR) (L-2011-021) approval to operate at a higher power level. Specifically, the activities associated with EC249969 replaced pneumatic level control equipment with digital level control equipment for the:

- Moisture Separator Reheaters (MSRs),
- High Pressure Feedwater Heaters (HP FWHs),
- Drain Collectors, in addition to
- Design of piping, power supply, cable/wiring and structural supports.

Procedure changes that fundamentally alter the existing means of performing or controlling design functions should be conservatively treated as adverse and should be evaluated under 10 CFR 50.59. Such changes include replacement of automatic action by manual action (or vice versa), analog to digital upgrades, changing a valve from "locked closed" to "administratively closed," and similar changes. Since EC249969 is a digital upgrade that fundamentally alters the existing means of performing or controlling design functions, the proposed activity is conservatively treated as adverse, a 10CFR 50.59 evaluation was performed.

The 50.59 Screening Process, which included a review of the Design Change Checklist and a failure modes and effects analysis (FMEA), identified a proposed activity requiring a 50.59 evaluation. An evaluation of the proposed activities produced the following conclusions:

The proposed activity uses new control systems technology that introduces new failure modes and effects. Foundation Fieldbus communications with Control In Field (CIF) technology is being introduced to replace the existing pneumatic level controllers, Guided Wave Radar technology is being introduced to measure vessel condensate levels, and Fieldbus capable Digital Valve Controllers are being installed to replace pneumatic valve positioners. The replacement instruments and respective bridles introduce several new possible failure modes. The most notable new failure modes are communications failures and electronic device failures (including loss of power). For each of the new devise failure modes, a backup system is available to limit the worst case effects to a reduction in plant operating efficiency. Although the FMEA identified several new failure modes, it shows that there are backup controls to limit the adverse effects of each type of device failure. The replacement MSRs, FWHs, and Drain Collectors instruments and bridles will not adversely affect the ability of the Heater Drain and Vent System to function within its current design requirements. This modification serves to improve overall plant reliability by replacing obsolescent equipment with new proven technology. Therefore, the proposed activity is considered neutral to nuclear safety under the EPU configuration, and a net benefit under the current plant configuration.

The primary facts that support this conclusion are detailed below.

- Cyber Security Program - The Heater Drain Fieldbus control system was integrated into the PI Data Historian network. As a result, this new control system will take advantage of the

security measures that have been built into the PI Network. These measures include internal and external firewalls. The new Fieldbus equipment will communicate with the existing PI Historian using the new Fieldbus Interface. Modules (FIMs) added as a result of this EC via the Plant Data Network (PDN). The equipment added by this EC are not critical digital devices within the scope of the cyber security program. The modification has been reviewed for cyber security considerations and found acceptable. Therefore, the proposed activity is considered neutral to nuclear safety under the current plant and EPU configurations.

- Human Factors - The operators will have newly available data and control functionality available at the HMI screen which is being installed inside the MSR/FW Heater Drain Control Panel. Local manual control of the level control valves will now be available through the HMI screen. Level control loop tuning and monitoring will also be available at the HMI screen. Additionally, operators will have continuously recorded data available via the site data historian (PI). The availability of these new control and operator functions represents an improvement over current conditions. The availability of historical data will offer the opportunity to trend and analyze operating conditions that were previously unavailable. This provides an enhancement to the current condition and also represents an improvement. Therefore, the proposed activity is considered neutral to nuclear safety under the current plant and EPU configurations.

- Instrument Setpoints - The activity revised level switch setpoints for the replacement MSRs and HP FWHs. New setpoints for alarms and the digital valve controllers are required because higher capacity equipment for the Heater Drain and Vents system is being installed in anticipation of operation at EPU conditions. The Drain Collector vessels will not be changed for EPU. The setpoints associated with Drain Collector high levels will not change. Therefore, the proposed activity is considered neutral to nuclear safety under the current plant and EPU configurations.

- Software Quality Assurance – The proposed activity installed Fieldbus capable Digital Valve Controllers for the MSRs, HP FWHs and DCs which use software programs for system configuration, troubleshooting, and data collection. These programs have been added to the Master Software Index and the Software Quality Assurance requirements have been documented. All software control,verification and validation, and documentation have been prepared in accordance with IM-AA-101, "Software Quality Assurance Program" and IM-AA-202, Rev. 2, "Software Quality Assurance Process." The software aspects have been documented in accordance with FPL's Software Quality Assurance Program. Therefore, the proposed activity is considered neutral to nuclear safety under the current plant and EPU configurations.

- Electromagnetic and Radio-Frequency Interference, Surge or Electromagnetic Discharge - The new electronic equipment being introduced as a result of this plant modification has been tested and certified to be in compliance with International Electrotechnical Commission (IEC) industry standards for susceptibility to EMI/RFI.

The 10 CFR 50.59 Screening documents demonstrated that this activity does not require any changes to the Technical Specifications. The 10 CFR 50.59 Screening identified possible adverse effects to the functions of system(s) as discussed in the UFSAR; therefore, a 10 CFR 50.59 Evaluation was performed. The 10 CFR 50.59 Evaluation demonstrated that no prior NRC review and approval is required.

PLANT CHANGE/MODIFICATION EC249978

LEFM - Measurement Uncertainty Recapture

REVISION 2

Summary:

EC249978 replaced the existing Cameron Leading Edge Flow Meter (LEFM) System with the more accurate CheckPius model to support increased accuracy in main feedwater flow measurement, and prepares the plant for operation at Extended Power Uprate (EPU) conditions, as described in St. Lucie Unit 2 EPU Licensing Amendment Request (LAR) (L-2011-021). This supports a more accurate secondary calorimetric thermal power calculation. The scope of EC249978 activities includes installation or replacement of hardware necessary to improve the accuracy of main feedwater flow measurements, but specifically excludes justification for the EPU maximum thermal power limit, and specifically excludes activities to implement the Distributed Control System (DCS)/LEFM interface. EC249978 implemented:

- Installation of LEFM 2A/2B Metering Sections in Feedwater (FW) piping.
- Installation of power supply cabling for transmitters and Control Room Central Processing Units (CPUs).
- LEFM communication media provisions for integration with existing systems. Mounting of local transmitter enclosures at the Turbine Building Mezzanine.
- Installation of pressure transmitters and connection of Resistance Temperature Detectors (RTDs) for LEFM flow calculations.
- Installation of raceway, new cables, and data cabling to Instrument Racks and the Control Room.
- Installation of LEFM Central Processing Unit (CPU) cabinets in the Control Room.
- Removal and/or partial re-use of the existing LEFM System.

The LEFM System is not discussed in the Technical Specifications (TS) or Bases. The requirements for using the LEFM-supported calorimetric calculation in order to calibrate the Nuclear and AT Power Channels are contained in TS surveillance requirement 4.3.1.1. This TS Surveillance Requirement is not adversely affected and does not require any change due to this modification. The Unit 2 TS do not address the Feedwater Regulating System (FWRS) or the Low Power Feedwater Control System (LPFWCS) or any of the components associated with the systems being modified. The proposed changes will perform the same functions as the existing systems and there are no new interfaces established with any other plant systems that are included in the TS.

Therefore, the proposed modification does not adversely impact any TS or Bases described SSCs or their ability to perform described design functions and there are no changes to the Unit 2 TS as a result of this modification.

Under normal power consumption, the heat load generated by electronic devices and components in the two CPU cabinets will be 887.2 Btu/hr. The net heat loading change to the Control Room is an additional 498.22 Btu/hr. The additional heat loading is considered adverse; thus this aspect of EC249978 activities required a 10CFR 50.59 evaluation.

The 10 CFR 50.59 Screening documents demonstrated that this activity does not require any changes to the Technical Specifications. The 10 CFR 50.59 Screening identified possible adverse effects to the functions of system(s) as discussed in the UFSAR; therefore, a 10 CFR 50.59 Evaluation was performed. The 10 CFR 50.59 Evaluation demonstrated that no prior NRC review and approval was required.

PLANT CHANGE/MODIFICATION EC249981

Control Room AC Margin Improvement

Revision 2

Summary:

Modification EC249981 implemented activities that prepared the plant for operation at Extended Power Uprate (EPU) conditions, as described in St. Lucie EPU License Amendment Request (LAR) (L-2011-021). The EPU project upgraded the Control Room Air Conditioning System (CRACS) to enable operation of the air conditioning units at a maximum Component Cooling Water (CCW) supply temperature of 120 F. EC249981 was implemented during the SL2-20 outage. Specifically, EC249981 upgraded the CRAC System air conditioning units (HVA/ACC 3A, 3B and 3C) by:

- replacing the evaporator cooling coil,
- replacing the reciprocating compressor with a screw compressor with integral motor,
- replacing the tube and shell refrigerant condenser with a brazed plate condenser,
- replacing the existing analog refrigerant circuit controls with a digital control system,
- replacing the CCW flow control valve,
- replacing the refrigerant piping system including valves, filters, dryers, etc.,
- replacing the existing system refrigerant,
- modifying the CCW supply and return piping to the refrigerant condenser,
- modifying the electrical power supply to the CRAC skid,
- modifying the electrical power supply to Security Lighting Panel LP2-2A2,
- removing and modifying existing CCW pipe supports,
- Installation of Control Room AC Refrigerant Monitor YE-25-1, and
- installation of H&V Room Oxygen Monitor YE-25-2 and Remote Sensor YE-25-3.

No aspects of EC249981 were dependent upon NRC approval of the EPU LAR.

The 10CFR 50.59 screening evaluation identified the existence of a credible common mode failure in the digital control system software program that resulted in the CRACS not meeting single failure criteria. This condition could adversely affect an UFSAR described design function for the CRACS such that no single active failure coincident with a loss of offsite power can result in the loss of functional performance. Radiological Dose Assessments - UFSAR Chapter 15, "Accident Analysis" credits the operation of the Control Room Emergency Cleanup System (CRECS) and the CRACS for various design basis accident control room radiological dose assessments. The proposed activity could affect the results of these assessments since Screen Design Function 2 is adversely affected. Analog to Digital Upgrades procedure changes that fundamentally alter the existing means of performing or controlling design functions should be conservatively treated as adverse and should be evaluated under 10 CFR 50.59. Such changes include replacement of automatic action by manual action (or vice verse), analog to digital upgrades, changing a valve from "locked closed" to "administratively closed," and similar changes. Based upon the above guidance, Question 3 in the 50.59 Screening regarding, "...a change to a procedure that adversely affects how UFSAR described SCC design functions are performed or controlled" is checked YES, thus requiring this Evaluation to be performed.

A review of the Unit 2 Technical Specifications (TS) was performed to identify all requirements

associated with the CRACS. The review identified TS 3/4.7.7 as specifying the limiting condition of operation for the control room emergency air cleanup system which together with the CRACS is the Control Room Ventilation System that assures control room habitability. However, there are no required changes to the Technical Specifications as a result of this modification. The upgrade to the control system for the CRACS does not adversely affect any design functions of the CRACS as stated in the UFSAR. This modification in the aggregate is considered neutral to nuclear safety under the EPU configuration, and a net benefit under the current plant configuration. The capability to provide an operable control room air conditioning unit is maintained for control room habitability through the implementation of this modification. Technical Specification 3/4.7.7 is met with the new digitally controlled CRACS equipment.

The 10 CFR 50.59 Screening documents demonstrate that this activity does not require any changes to the Technical Specifications. The 10 CFR 50.59 Screening identified possible adverse effects to the functions of system(s) as discussed in the UFSAR; therefore, a 10 CFR 50.59 Evaluation was performed. The 10 CFR 50.59 Evaluation demonstrated that no prior NRC review and approval is required.

PLANT CHANGE/MODIFICATION EC271287

Diesel Oil Storage Tank Operating Margin Modification (On-line)

Revision 1

Summary:

Modification EC271287 supported the St. Lucie Unit 2 Extended Power Uprate (EPU) LAR (L-2011-021) approval to operate at a higher power level. EC271287 enables each of two Unit 2 Diesel Oil Storage Tanks (DOST 2A, DOST 2B) to be filled to a greater stored volume. Specifically, this modification:

- Installed loop seals in the overflow piping
- Added DOST building penetrations with pipe sleeve boots to allow vacuum breaker lines to terminate outside of the building
- Increased the thresholds of the DOST building exterior doorways
- Raised the DOST low level alarm setpoint

The physical dimensions of the DOSTs and DOST building were not changed. However, new exterior doorway steel plates was added to each of the DOST Building exterior doorways. EC271287 implementation did not require prior EPU LAR approval.

EC271287 provides additional diesel fuel storage capacity to the DOSTs. The additional fuel will provide increased margin to allow the emergency diesel generator to operate for an additional period of time between tank refills. The increased storage capability will increase the total weight that is supported by the tank foundations and their design. As such, the increased storage capability could adversely affect this UFSAR described design function. Thus, this aspect of EC271287 required evaluation pursuant to 10CFR 50.59

EC271287 increased the static and dynamic loading for the DOSTs that was previously considered in the seismic design due to the additional weight of the fuel oil, overflow piping and fittings. As such, the increased storage capability could adversely affect these UFSAR described design functions. Thus, this aspect of EC271287 required evaluation pursuant to 10CFR 50.59.

UFSAR changes to the Fire Hazard Analysis for Fire Area AA and BB discussed in Appendix 9.5A, were identified in the EC. The existing secondary containment will hold diesel oil up to a height of 10'-6" (i.e. the height of the DOST Building doors). Based upon the additional fuel oil storage capacity, Calculation 25486- 266-CYC-0001-00001 has determined a new spill height for the DOST Building. The new maximum volume of oil is 45,004 gallons, which could fill the secondary containment to a height of 11'-00" in the event of a failure of the DOST or associated piping. Modifications to the DOST Building to accommodate for the increased spill height are documented in Section 2.3.1.1 of EC271287. The modification to the DOST Building doorways will not adversely affect the DOST Building. Indoor containments are sized for 100% of the tank volume. As such, in the event of a failure of the DOST or associated piping, the increased storage volume adversely affects the existing DOST Building secondary spill containment design capacity. Based upon the above, the changes proposed by EC271287 that adversely affect a UFSAR described design function are:

1. The additional fuel capacity increases the static and dynamic (sloshing) loads on the seismically designed tank structure and its foundation.
2. The additional fuel oil capacity increases the secondary containment volume requirement in the event of a failure of the DOST or associated piping.

The increase in the DOST's maximum storage capability will increase the total weight that is transferred to the DOST foundations. The DOST foundations were designed to support a tank completely filled with water. Water is denser than diesel fuel, and 2-inches of freeboard will be maintained at the maximum tank level. Therefore the existing foundations can support the additional volume of diesel fuel. The effect of the increased level of oil was shown to be acceptable for design basis seismic events. The increase in fuel oil storage capacity has no adverse effect on the ability of the DOSTs to withstand design basis earthquake loads. Calculation confirms that the new overflow piping and loop seals do not introduce a Seismic II/I concern. With the increase in oil storage volume, the DOST Building secondary containment capacity will be increased. New exterior doorway steel plates were added to each of the DOST Building exterior doorways at the 10'-6' elevation. These steel plates were sealed to the building walls with a gasket such that the DOST Building is able to contain diesel fuel up to a height of 11'-0". As a result this change is considered neutral to nuclear safety under the EPU configuration, and a net benefit under the current plant configuration.

The 10 CFR 50.59 Screening documents demonstrate that this activity does not require any changes to the Technical Specifications. The 10 CFR 50.59 Screening identified possible adverse effects to the functions of system(s) as discussed in the UFSAR; therefore, a 10 CFR 50.59 Evaluation was performed. The 10 CFR 50.59 Evaluation demonstrated that no prior NRC review and approval is required.

PLANT CHANGE/MODIFICATION EC275895

Permanent Removal of St. Lucie Unit 2 RCP 2B1 Whip (Cable) Restraints

Revision 0

EC275895 replaced of RCP 2B1 rotating assembly and removed the upper cable (whip) restraint, as detailed in items 1 and 2.

1. In conjunction with the RCP Motor Refurbishment Project it has been decided to replace the 2B1 RCP rotating assembly and associated parts during SL2-20. A new rotating assembly is being purchased and will be used to replace the existing rotating assembly. The new assembly has been upgraded with original equipment manufacturer (OEM) enhancements. The enhancements include new materials of construction, a new pump to motor coupling (Curvic Coupling), and a new vibration probe mounting bracket.

2. To allow for future maintenance of the reactor coolant pump the upper RCP cable (whip) restraints shall also be permanently removed. In addition to removing an interference that impedes the disassembly of the pump it will also reduce the radiological dose required to reinstall the two 4-inch cables.

The installation of new RCP 2B1 internals is considered an equivalent replacement that does not alter the design functions of the RCP or RCS. The new cover and impeller were designed to the same requirements as the original cover and impeller and are equivalent to the original parts in design and materials. Other design changes to internal parts are being made to improve reliability and eliminate potential failure mechanisms. The new materials are internal to the pump and with the exception of the cover are not part of the pressure boundary and therefore will not affect the RCS pressure boundary design or function. The reactor coolant pumps will still deliver the required flow at the given design operating pressure and temperature.

Based on the above, the replacement of RCP 2B1 Rotating Assembly, including the Curvic Coupling and vibration probe bracket, is considered equivalent to the original design and does not constitute a change to an SSC that adversely affects a UFSAR design function; therefore, this portion of the activity screened out and did not require a 10CFR50.59 evaluation.

The design function of RCP 2B1 whip restraints is described in UFSAR Section 3.6. Permanent removal of the upper whip restraints on RCP 2B1 pump casing constitutes a change to an SSC that adversely affects a UFSAR described design function. Therefore, with respect to the removal of the upper whip restraints on RCP 2B1 pump casing, considerations regarding prior NRC approval were addressed under a 10CFR50.59 evaluation.

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 3/4.4.6.1 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.

A License Amendment Request was not required. 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.

PLANT CHANGE/MODIFICATION EC249666

Permanent Removal of St. Lucie Unit 2 RCP 2A1 Whip (Cable) Restraints

Revision 0

EC275666 replaced of RCP 2A1 rotating assembly and removed the upper cable (whip) restraint, as detailed in items 1 and 2.

1. In conjunction with the RCP Motor Refurbishment Project it has been decided to replace the 2A1 RCP rotating assembly and associated parts during SL2-20. A new rotating assembly is being purchased and will be used to replace the existing rotating assembly. The new assembly has been upgraded with original equipment manufacturer (OEM) enhancements. The enhancements include new materials of construction, a new pump to motor coupling (Curvic Coupling), and a new vibration probe mounting bracket.

3. To allow for future maintenance of the reactor coolant pump the upper RCP cable (whip) restraints shall also be permanently removed. In addition to removing an interference that impedes the disassembly of the pump it will also reduce the radiological dose required to reinstall the two 4-inch cables.

The installation of new RCP 2A1 internals is considered an equivalent replacement that does not alter the design functions of the RCP or RCS. The new cover and impeller were designed to the same requirements as the original cover and impeller and are equivalent to the original parts in design and materials. Other design changes to internal parts are being made to improve reliability and eliminate potential failure mechanisms. The new materials are internal to the pump and with the exception of the cover are not part of the pressure boundary and therefore will not affect the RCS pressure boundary design or function. The reactor coolant pumps will still deliver the required flow at the given design operating pressure and temperature.

Based on the above, the replacement of RCP 2A1 Rotating Assembly, including the Curvic Coupling and vibration probe bracket, is considered equivalent to the original design and does not constitute a change to an SSC that adversely affects a UFSAR design function; therefore, this portion of the activity screened out and did not require a 10CFR50.59 evaluation.

The design function of RCP 2A1 whip restraints is described in UFSAR Section 3.6. Permanent removal of the upper whip restraints on RCP 2A1 pump casing constitutes a change to an SSC that adversely affects a UFSAR described design function. Therefore, with respect to the removal of the upper whip restraints on RCP 2A1 pump casing, considerations regarding prior NRC approval were addressed under a 10CFR50.59 evaluation.

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 3/4.4.6.1 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.

A License Amendment Request was not required. 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.

PLANT CHANGE/MODIFICATION EC277173

2B1 Safety Injection Nozzle Thermal Sleeve Dislodged 2A1 Safety Injection Nozzle Thermal Sleeve Rotated, CSB And Surveillance Capsule Holders Damaged

Rev. 0

Summary:

A cold leg safety injection nozzle thermal sleeve was found between the flow baffle skirt and reactor vessel wall while performing a Foreign Object Search and Retrieval (FOSAR) of the reactor vessel after core barrel removal. Further investigation revealed the source to be 2B1 cold leg safety injection nozzle thermal sleeve. Subsequent inspection found the 2A1 safety injection nozzle thermal sleeve rotated from its design position, as well as damage to two reactor vessel material surveillance capsule holders, the core support barrel and CSB Snubber Blocks apparently caused by the dislodged thermal sleeve.

Per Westinghouse Letter LTR-RIDA-12-158, Rev. 1, "Operating experience has shown that nozzle thermal sleeves can come loose, dislodge, and migrate through parts of the RCS; however, operating experience has also shown that no significant damage to the RCS systems will result. Furthermore, operating experience has shown that only thermal sleeves in the SI nozzles of various CE design plants have come loose, and in some cases have loosened to the point of becoming dislodged and carried through parts of the system."

EC277173 documents the Use-As-Is disposition for ARs 01795296, (Debris Found Between the Flow Baffle Skirt and Vessel Wall (determined to be 2B1 thermal sleeve)), 1795649 (U2 RV, Degraded Surveillance Holders Require Evaluation), 1796008 (U2 Core Support Barrel Degraded Condition Requires Evaluation), 1796227 (Core Support Barrel Support Snubber Block Deficiencies), 1797638 (2A1 SI Thermal Sleeve), and 1798633 (2B1 SI Nozzle Cladding Degradation). As documented in the 10 CFR 50.59 screening of this EC, certain aspects of the proposed change are considered adverse and require a 10 CFR 50.59 evaluation.

The aspects of this EC subject to this 10 CFR 50.59 evaluation are:

- Analysis of the SI cold leg nozzles due to the adverse effect of the loss of the thermal sleeve on the piping analyses;
- Analysis of the potential for a dislodged SI cold leg nozzle thermal sleeve to become a loose part and impact other components;
- Analysis of the Core Support Barrel (CSB) due to the adverse effect of the wear damage from interaction with the thermal sleeve.

Therefore, the 10CFR50.59 evaluated the design configuration of a missing thermal sleeve, one thermal sleeve being in a rotated position and damage to the CSB. The potential for a thermal sleeve becoming a loose part is addressed in the malfunction questions in the 10CFR50.59. If a thermal sleeve becomes a loose part, the plant will either shutdown or an operability evaluation will be required at that time.

This EC accepts for use as-is, the conditions of the safety injection (SI) system piping nozzle thermal sleeves located at the reactor coolant system (RCS) cold legs. The Technical Specifications were reviewed for impact, including Sections 2.2, Limiting Safety System Settings, 3/4.2, Power Distribution Limits, 3/4.4, Reactor Coolant System, 3/4.5, Emergency Core Cooling Systems (ECCS), and 5.7, Component Cyclic or Transient Limits. Technical Specification 3.2.5 which specifies the RCS minimum total flow rate was also reviewed for impact. The following Technical Specifications related to SI (ECCS) and Shutdown Cooling were also reviewed for impact; 3.5.1, 3.5.2, 3.1.2.3, 3.9.8.1, 3.9.8.2. No Technical Specification changes are required to implement this EC.

Technical Specification operability of the both the RCS and the SI system (or emergency core cooling system, ECCS) is not impacted by this change. The ability of the RCS and ECCS to comply with existing Technical Specification requirements, including flow requirements, is not impacted by this EC.

Reactor Vessel Surveillance Capsule Holder

This EC accepts for use as-is, the conditions of the degraded surveillance capsule holders at the 83° and 97° locations. Technical Specification Surveillance 4.4.9.1.2 requires that specimens be removed and examined as required by 10 CFR 50 Appendix H. Acceptance of the degraded surveillance capsule holders does not impact the ability to remove and test specimens. No change is required for this Technical Specification requirement. Note that UFSAR Table 5.3-9, Capsule Assembly Removal Schedule indicates that the capsule assemblies at 83° and 263° have been removed; the capsule assemblies at 97°, 104°, 277°, and 284° remain in place.

Core Support Barrel

This EC accepts for use as-is, the conditions of the damaged core support barrel (CSB) and CSB Snubber Lugs. There are no Technical Specifications associated with the CSB.

RTD Thermowells and Instruments

If the dislodged SI thermal sleeve hits the RTD thermowell (due to reverse flow though an idle RCP) under a point load or distributed load condition, the thermowell will bend but not collapse, i.e., the thermowell will not break. Therefore, the RCS pressure boundary would be maintained. However, the operation of the RTD would likely be impacted. Technical Specification 3.2.5 requires the DNB-related parameter of Cold Leg Temperature (as shown on Table 3.2-2 of the COLR) to be maintained. Damage to a RTD Thermowell and the associated Instrument could affect the ability of the instrument to read the Cold Leg Temperature. However, the RTD Instruments are redundant, so that loss of one instrument would not require shutdown.

SI Nozzle Thermal Sleeves (including impact on RCS and SI flowrates)

This EC accepts for use as-is, the conditions of the safety injection (SI) system piping nozzle thermal sleeves located at the reactor coolant system (RCS) cold legs. The Technical Specifications were reviewed for impact, including Sections 2.2, Limiting Safety System Settings, 3/4.2, Power Distribution Limits, 3/4.4, Reactor Coolant System, 3/4.5, Emergency Core Cooling Systems (ECCS), and 5.7, Component Cyclic or Transient Limits. Technical Specification 3.2.5 which specifies the RCS minimum total flow rate was also reviewed for impact. The following Technical Specifications related to SI (ECCS) and Shutdown Cooling were also reviewed for impact; 3.5.1, 3.5.2, 3.1.2.3, 3.9.8.1, 3.9.8.2. No Technical Specification changes are required to implement this EC.

PLANT CHANGE/MODIFICATION EC277805

Increase in RTD Hot Leg Response Time

Rev. 0

Summary:

The RCS hot leg RTD signals are provided to the reactor protection system (RPS) to calculate thermal power, which affects the calculation of the thermal margin/low pressure (TM/LP) trip, the local power density (LPD) trip and the variable high power (VHP) trip. A change in the maximum response time value assumed in the safety analysis has the potential of negatively affecting the results of the events where these three reactor trips are credited for mitigation. The accidents of interest where these reactor trips are credited include Feedwater Malfunction (hot full power cases only), Pre-Trip Hot Full Power Steamline Break, RCS Depressurization, CEA (Rod) Withdrawal at Power, CEA (Rod) Drop, CEA (Rod) Ejection, SGTR, and CEA (Rod) Withdrawal from Subcritical.

The SSCs related to this evaluation include the hot leg RTDs which provide input data to the reactor protection system. The design functions for this system and components provided in chapter 7 of the UFSAR. The 10CFR 50.59 screen for this activity determined that the design function could be adversely affected by the changes in response time from 8 to 15 seconds, as the margins of safety in the Chapter 15 analyses could be altered by the proposed activity. Therefore, the proposed modification to increase the response time for these detectors for Cycle 20 only, which will be documented in UFSAR Table 13.7.2-1, was evaluated under the 50.59 rule.

The Technical Specifications were reviewed for this activity. The only relevant specification for this activity is 3/4.3.1, Reactor Protective Instrumentation. The change in response time for the RCS hot leg RTDs does not have any effect on this Technical Specification, as response times are not specified in the Tech Specs.

Based on the discussion provided above, no License Amendment was required for this activity.

SECTION 2

50.59 EVALUATIONS

Engineering Evaluation (PSL-ENG-SEMJ-09-055) -Zinc Addition to the Reactor Coolant System on PSL Unit 2

The activity which was evaluated is the addition of zinc acetate to PSL Unit 2 RCS as a means of reducing dose rates and the potential for Primary Water Stress Corrosion Cracking (PWSCC). This evaluation addresses the changes to RCS chemistry associated with a zinc addition program and the effects on components within the RCS and supporting systems including fuel assemblies that will come in contact with the zinc acetate.

This 1OCFR50.59 Evaluation supports the change in the Revision 1 of the Engineering Evaluation 10CFR50.59 Screen which "screened-in" as a result of a change to the justification of Question 5.

PSL Unit 2 Technical Specification 3.4.7 addresses RCS chemistry parameters and limits for "Dissolved Oxygen", "Chloride" and "Fluoride" but does not address other chemicals or additives injected into the RCS to maintain pH or other desired parameters. Since the addition of zinc will not affect the concentrations of oxygen, chlorides or fluorides and is in the same category as lithium hydroxide, hydrazine, etc.; therefore. no change to Technical Specification 3.4.7 is required.

Technical Specification 3.4.8 addresses the specific activity of primary coolant. Subparagraph 'a' specifies limits of activity in the reactor coolant in relation to Dose Equivalent Iodine-131, which is not applicable to zinc addition. Subparagraph 'b' specifies limits for radio nuclides in the reactor coolant but does not specify limits for individual radio nuclides. The only limit is given in terms of the average sum (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of the sampling) of the average beta and gamma energies per disintegration. Isotopes, other than isotopes of iodine, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant, are considered. This is called $\bar{E}$ or the average disintegration energy. The average value is limited to $100/\bar{E}$ μCi/gram. According to Westinghouse "LTR-CDME-09-29", the typical design value of $\bar{E}$ for all nuclides considered is about 0.31, thus giving the coolant a mean value of 323 μCi/gram. Westinghouse calculated Post-Zinc Reactor Coolant Activity Concentrations for Radiocobalt activities that will bound the maximum expected increases based on data from other zinc plants. From Westinghouse's evaluation, the maximum activity anticipated due to zinc addition is 3.36X10-2 μCi/gram at ambient conditions. This value is orders of magnitude lower than the 323 μCi/gram derived from the average disintegration energies; therefore, the addition of zinc to PSL Unit 2 will not have a significant effect on coolant radionuclide concentrations or imposed limits. Consequently, there are no changes required to the specific activity in Technical Specification 3.4.8.

Based on the 10CFR 50.59 evaluation, UFSAR, DBD and other references, prior NRC approval and a License Amendment is not required to implement a zinc addition program at PSL Unit 2.

EC277297 PSL-ENG-SEMS-10-022

<u>The Effects of Extended Power Uprate on Emergency Diesel Generator Loads</u>

Rev. 0

Summary:

EC277297 implemented changes to an engineering evaluation and two calculations that support the design basis for Emergency Diesel Generator (EDG) loading within certain individual time frames designated as Load Blocks and fuel consumption due to St. Lucie Unit 2 Extended Power Uprate (EPU) modifications. An UFSAR change is also performed with the output of these documents.

EPU modifications affecting EDG loading are:

- Engineering Change EC249981 "Control Room Air Conditioning Margin Improvement" – This modification replaces the air conditioner compressors and control transformers that are powered by the EDG. It retains the air conditioner ventilation fans that are powered by the EDG. The air condition compressor load is reduced, the control transformer load is new(added) and the fan load remains the same.

- Engineering Change EC277248 "Control Room Air Conditioning Control Circuit Change" – This modification revised the control scheme for the air conditioner components resulting in different EDG load block timing for the air conditioner compressors, control transformers and fans.
- Engineering Change EC 246487 "TCW Heat Exchanger Replacement" – This modification replaced the TCW heat exchanger and tube side flow orifice that resulted in increased flow for the Intake Cooling Water (ICW) pumps and increased EDG load.

- Engineering Change EC 249990 "Unit 2 CS Pump Flow Limitation" – This modification installed flow restricting orifices in the discharge of the Containment Spray (CS) pumps. Though the calculated maximum CS flow was reduced, separate analyses referenced in EC249990 and discussed in Calculation PSL-2FSM-10-008 (revised per this EC-EVAL) conclude that system conditions result in higher CS flows. Thus, EDG load is increased as a result.

EC277297 does not involve any activity that performs a physical change to any PSL-2 SSC's.

The screening evaluation resulted in the following activity screening in. This activity is summarized below and are addressed in the 10CFR 50.59 Evaluation.

EDG Load Margin: The additional loads due to the cumulative effects of the EPU modifications resulted in a reduction in the margin between the peak steady state load and the EDG rating. The highest Load Block (#12) steady state load rises from 3232KW to 3328.2KW as shown on UFSAR Figure 8.3-4 as changed under this EC. This is a change from 87.7% of EDG Rating of 3686KW to 90.3% of EDG rating. The steady state loads in the preceding and succeeding Load Blocks are 3233.5KW and 3256.3KW, respectively. These values represent 87.75% and 88.4% of EDG Rating respectively. Though UFSAR Figure 8.3-4 is not referenced in UFSAR Section 8.1.1.2.m where the margin requirement is stated, this figure represents the values for cumulative calculated load consumption calculated in Engineering Evaluation PSL-ENG-SEMS-10-022, Rev. 1. This margin reduction constitutes an adverse effect on an UFSAR design function

TS 3.8.1.1 and 3.8.1.2 as revised by License Amendment 163 were reviewed for impact from the revision to EDG loading provided in this evaluation. No changes to the Technical Specifications were required.

The revision to the evaluated EDG loads have a cumulative effect, whereby the margin between the EDG loads and EDG rating is reduced during certain Load Blocks, following Design Basis Accidents. These margin reductions are not significant considering the greater than 7 day required operating time for the EDG following a Design Basis Accident. Functions described in the safety analysis for the system are maintained and thus plant safety is unaffected.

The 10CFR 50.59 evaluation concluded that no LAR per 10CFR 50.90 was required.