ATTACHMENT 1
LICENSE AMENDMENT REQUEST
EXTENDED POWER UPRATE

DESCRIPTIONS AND TECHNICAL JUSTIFICATIONS
FOR THE
RENEWED FACILITY OPERATING LICENSE,
TECHNICAL SPECIFICATIONS,
AND
LICENSING BASIS CHANGES

FLORIDA POWER & LIGHT
ST. LUCIE UNIT 2

This coversheet plus 45 pages

DESCRIPTIONS AND TECHNICAL JUSTIFICATIONS FOR THE RENEWED FACILITY OPERATING LICENSE, TECHNICAL SPECIFICATIONS, AND LICENSING BASIS CHANGES

1.0 DESCRIPTION

Florida Power & Light Company (FPL) is proposing to amend the Renewed Facility Operating License (RFOL) No. NPF-16 for St. Lucie Nuclear Plant Unit No. 2 (St. Lucie Unit 2).

The proposed license amendment request (LAR) will revise the RFOL to permit St. Lucie Unit 2 to operate at a maximum steady-state reactor core thermal power of 3020 megawatts thermal (MWt). The requested increase constitutes an extended power uprate (EPU) and includes a measurement uncertainty recapture (MUR) uprate. The proposed amendment is requested to provide greater unit generating power. FPL developed this LAR consistent with the guidance provided in NRC Review Standard RS-001, Review Standard for Extended Power Uprate (Reference 1) and NRC Regulatory Issue Summary (RIS) 2002-03, Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications (Reference 2). Power ascension to EPU power level is planned following NRC approval of the EPU LAR and completion of the required plant modifications. There are no other St. Lucie Unit 2 LARs that are required for implementation of the EPU.

FPL has reviewed the RFOL, technical specifications (TS), and current licensing basis (CLB) and has determined that no revisions to these documents other than those addressed in this LAR are required to properly control plant operations and configuration under EPU conditions. Markups of the proposed RFOL and TS are provided in Attachment 3. To support the MUR portion of the uprate, FPL is installing new feedwater flow instrumentation, specifically, the Cameron Leading Edge Flow Measurement (LEFM) CheckPlus™ System. The allowed outage times and required actions for the new feedwater flow measurement instrumentation will be incorporated into the Updated Final Safety Analysis Report (UFSAR). Note that the proposed markups of the TS Bases and Core Operating Limits Report (COLR) are provided for information only in Attachments 4 and 8, respectively. In addition, the regulatory commitments identified in Attachment 7 are required to be completed as stated or prior to implementation of the EPU.

In addition to the changes to support the EPU, this LAR includes changes that are not required for EPU, but are affected by EPU changes and provide enhancements to the St. Lucie Unit 2 TS. Changes that are not required to support EPU are identified as such.

2.0 BACKGROUND

St. Lucie Unit 2 is currently licensed to operate at a core thermal power of 2700 MWt. Approval of this LAR would authorize FPL to operate the unit at a core thermal power of 3020 MWt. This represents a net increase in licensed thermal power of approximately 11.85 percent and includes a 10.0 percent power uprate and a 1.7 percent measurement uncertainty recapture. The net increase is calculated as follows:

$$(2700 \text{ MWt} \times 1.10) \times 1.017 \cong 3020 \text{ MWt}$$

$$(3020 \text{ MWt} - 2700 \text{ MWt}) / 2700 \text{ MWt} \cong 11.85\%$$

Due to the magnitude of this increase in licensed thermal power, this power uprate is defined as an EPU.

Like most nuclear units, St. Lucie Unit 2 was originally designed with feedwater flow instrumentation and analytical techniques that were appropriate at the time. Since then, improvements have occurred in feedwater flow measurement instrumentation and associated power calorimetric uncertainty values. Based on the installation of new LEFM feedwater flow instrumentation and the associated reactor core power uncertainty values, FPL is proposing a 1.7 percent MUR uprate.

FPL has evaluated the impact of the power uprate for the applicable systems, structures, and components (SSC), and the safety analyses at St. Lucie Unit 2. The results of this evaluation are described in LAR Attachment 5, Licensing Report (LR). The EPU LR supports the requested EPU changes to the RFOL, TS, and CLB. The LR also provides a description of the plant modifications associated with the EPU. The LR works in concert with the other attachments to this LAR to provide a comprehensive evaluation of the effects of the proposed uprate.

3.0 PROPOSED CHANGES

The requested changes involve revisions to the RFOL, TS, and CLB. There are no other St. Lucie Unit 2 LARs that are required to implement the EPU. Changes that are not required for EPU, but included in this LAR, are specified as such below.

FPL has reviewed the RFOL, TS, and CLB and has determined that no revisions other than those addressed in this LAR are required to properly control plant operations and configuration under EPU conditions.

3.1 Renewed Facility Operating License and Technical Specifications Changes

1. RFOL NO. NPF-16, CONDITION 3.A - MAXIMUM POWER LEVEL

- The maximum reactor core power level is revised from “2700 megawatts (thermal)” to “3020 megawatts (thermal).”
- Outdated information regarding restrictions on maximum reactor core power that were in place prior to replacement of the steam generators is deleted.

<u>Basis for the Change:</u> The results of the analyses and evaluations performed and discussed in this LAR demonstrate that the proposed increase in licensed core thermal power can be safely and acceptably achieved by satisfying all applicable acceptance criteria, provided the regulatory commitments in Attachment 7 are completed as stated. The deleted material concerned restrictions on core power level that were in place before the replacement of the steam generators (SGs).

2. INDEX

DEFINITIONS

- “1.11 $\bar{E}$ - AVERAGE DISINTEGRATION ENERGY” is changed to “1.11 DOSE EQUIVALENT XE-133” and

- "1.16 LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE" is changed to "1.16 DELETED."

LIST OF FIGURES

- "3.1-1MINIMUM BORIC ACID STORAGE TANK VOLUME AS A FUNCTION OF STORED BORIC ACID CONCENTRATION" is changed to "MINIMUM BAMT VOLUME VS STORED BORIC ACID CONCENTRATION"
- "3.4-1 DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMITS VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1µCi/GRAM DOSE EQUIVALENT I-131" is changed to "3.4-1 DELETED"
- "3.4-2 ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 55 EFPY, HEATUP, CORE CRITICAL, AND INSERVICE TEST" is changed to "3.4-2 ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 47 EFPY, HEATUP, CORE CRITICAL, AND INSERVICE TEST"
- "3.4-3 ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 55 EFPY, COOLDOWN AND INSERVICE TEST" is changed to "3.4-3 ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 47 EFPY, COOLDOWN AND INSERVICE TEST"

LIST OF TABLES

- "3.2-2 DNB MARGIN LIMITS" is changed to "3.2-2 DELETED"

Basis for the Change: These are conforming changes to ensure that the TS Index correctly reflects the applicable TS.

3. TS 1.11, DEFINITIONS - $\bar{E}$ AVERAGE DISINTEGRATION ENERGY

- The definition $\bar{E}$ "- AVERAGE DISINTEGRATION ENERGY" is deleted and replaced with the definition "DOSE EQUIVALENT XE-133" provided below:

  DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (µCi/gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

Licensing Report: Section 2.9.2, Radiological Consequences for Alternative Source Term

Basis for the Change: The change from Average Disintegration Energy ($\bar{E}$) to Dose Equivalent Xe-133 supports the EPU analyses determination of dose consequences. The definition for Dose Equivalent Xe-133 is similar to the definition for Dose Equivalent I-131. The determination of Dose Equivalent Xe-133 will be performed in a manner similar to that currently used in determining Dose Equivalent I-131, except that the calculation of Dose

Equivalent Xe-133 is based on the acute dose to the whole body and considers the noble gases Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138, which are significant in terms of contribution to whole body dose. Some noble gas isotopes are not included due to low concentration, short half life, or small dose conversion factor.

When the $\bar{E}$ is determined using a design basis approach, it is assumed that 1.0 percent of the power is being generated by fuel rods having cladding defects and there is no removal of fission gases from the letdown flow. The value of $\bar{E}$ using this approach is dominated by Xe-133. The other nuclides have relatively small contributions. However, during normal plant operation, there are typically only a small amount of fuel cladding defects and the radioactive nuclide inventory can become dominated by tritium and corrosion/activation products, resulting in the determination of a value of $\bar{E}$ that is very different than what would be calculated using the design basis approach. Because of this difference, the accident dose analyses become disconnected from plant operation and the limiting condition for operation (LCO) becomes essentially meaningless. This results in a TS limit that can vary during operation as different values for $\bar{E}$ are determined.

The current $\bar{E}$ definition includes radioisotopes that decay by the emission of both gamma and beta radiation. This change will implement an LCO that is consistent with the whole body radiological consequence analyses, which are sensitive to the noble gas activity in the primary coolant, but not to other non-gaseous activity currently captured in the $\bar{E}$ definition. LCO 3.4.8 currently specifies the limit for primary coolant gross specific activity as $100/\bar{E}$ microcuries per gram (µCi/gram).

This change incorporates Dose Equivalent Xe-133 defined in TS 1.11, using the Environmental Protection Agency Federal Guidance Report No. 12 (FGR 12), External Exposure to Radionuclides in Air, Water, and Soil (Reference 3), as the source of whole body dose conversion factors. The change is acceptable from a radiological dose perspective, since it will result in an LCO that more closely relates the non-iodine reactor coolant system (RCS) activity limits to the dose consequence analyses which form their bases. The change is consistent with the dose conversion factors used in the applicable dose consequence analyses. This definition change is consistent with Technical Specification Task Force (TSTF), TSTF-490, Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec (Reference 4).

4. TS 1.16, DEFINITIONS - LOW TEMPERATURE OVERPRESSURE PROTECTION RANGE -RCS

- The definition "LOW TEMPERATURE OVERPRESSURE PROTECTION RANGE - RCS" is deleted.

<u>Basis for the Change:</u> The deletion of this definition is administrative since the actual values for the low temperature overpressure protection (LTOP) settings are provided in LCO 3.4.9.3 Reactor Coolant System – Overpressure Protection Systems. This change is consistent with NUREG-1432, Standard Technical Specifications Combustion Engineering Plants (Reference 5), which does not contain a definition for Low Temperature Overpressure Protection Range - RCS.

5. TS 1.25, DEFINITIONS - RATED THERMAL POWER

- The definition "RATED THERMAL POWER" is revised from "2700 MWt" to "3020 MWt."

Basis for the Change: The results of the analyses and evaluations performed and discussed in the LR sections and appendices and the other LAR attachments demonstrate that the proposed increase in licensed core thermal power can be safely and acceptably achieved by satisfying all applicable acceptance criteria. This includes fulfillment of the regulatory commitments contained in Attachment 7.

6. TS 2.1, SAFETY LIMITS – FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES FOUR REACTOR COOLANT PUMPS OPERATING

- "FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES FOUR REACTOR COOLANT PUMPS OPERATING" is replaced.

Licensing Report: Sections 1.1, Nuclear Steam Supply System (NSSS) Parameters and 2.8.5.0 Accident and Transient Analyses

Basis for the Change: The restrictions of the reactor core thermal margin safety limit prevent overheating of the fuel cladding and possible cladding deformation which could result in the release of fission products to the reactor coolant. Overheating is prevented by maintaining the steady state peak linear heat rate below the level at which centerline fuel melting will occur by restricting fuel operation to within the nucleate boiling regime where heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. The curves of TS Figure 2.1-1 show the loci of points of thermal power, RCS pressure, and vessel inlet temperature with four reactor coolant pumps (RCPs) operating for which the departure from nucleate boiling ratio (DNBR) limit is not violated. A revised TS Figure 2.2-1 is provided to reflect the EPU analysis.

7. TS 2.2, LIMITING SAFETY SYSTEM SETTINGS – TABLE 2.2-1 – REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

- Table 2.2-1, Functional Unit 6, Steam Generator Pressure –Low, Note (2) on the Trip Setpoint and Allowable Value are changed to superscripts
- Table 2.2-1, Functional Unit 8, "Steam Generator Level – Low" is changed to "Steam Generator Level– Low $^{(6), (7)}$" to include two new notes as described below,
- Table 2.2-1, Functional Unit 8, Steam Generator Level - Low, the Trip Setpoint is changed from "20.5%(3)" to "35.0%$^{(3)}$",
- Table 2.2-1, Functional Unit 8, Steam Generator Level - Low, the Allowable Value is changed from "19.5%(3)" to "34.1%$^{(3)}$",
- Table 2.2-1, Functional Unit 14, Reactor Coolant Flow - Low, in the Trip Setpoint column "95.4% of design Reactor Coolant flow with four pumps operating*" is changed to "95.4% of minimum Reactor Coolant flow with four pumps operating*",
- Table 2.2-1, Functional Unit 14, Reactor Coolant Flow-Low, in the Allowable Value column "94.9% of design Reactor Coolant flow with four pumps operating*" is changed to "94.9% of minimum Reactor Coolant flow with four pumps operating*",

- Table 2.2-1, Functional Unit 14, Reactor Coolant Flow - Low, the current footnote is changed from "* Design reactor coolant flow with four pumps operating is the minimum RCS flow specified in the COLR Table 3.2-2." to "* For minimum reactor coolant flow with four pumps operating, refer to COLR Table 3.2-2."
- Table 2.2-1, TABLE NOTATION, Notes (6) and (7) are added as follows:

"(6) If the as-found channel setpoint is either outside its predefined as-found acceptance criteria band or is not conservative with respect to the Allowable Value, then the channel shall be declared inoperable and shall be evaluated to verify that it is functioning as required before returning the channel to service."

"(7)The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Trip Setpoint or a value that is more conservative than the Trip Setpoint, otherwise that channel shall not be returned to OPERABLE status. The Trip Setpoint and the methodology used to determine the Trip Setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in the UFSAR."

Licensing Report: Section 1.1, Nuclear Steam Supply System (NSSS) Parameters, Section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems, Section 2.4.2, Plant Operability, Section 2.13, Risk Evaluation and Appendix E Supplement to Licensing Report Section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems

Basis for the Change: The reactor trip setpoints are the values at which the reactor trips are set for each parameter. The trip values have been selected to ensure that the reactor core and RCS are prevented from exceeding their safety limits.

The accidents and transient analyses used the conservative pre-EPU TS value of 20.5 percent for reactor trip on low SG level and have shown that the results are within acceptable safety limits. Hence, this TS change is not related to the EPU increase in power level but is based on additional risk reduction following postulated total loss of feedwater flow events.

Risk evaluations identified that the additional post reactor trip decay heat increase from the EPU power level would result in reduction in operator action times for successfully responding to total loss of feedwater scenarios, and potential operator action times for initiating once through cooling in the event that feedwater restoration was not achieved. It was determined that, with the reactor trip on low SG level raised to 35.0 percent and revision of plant procedures that provide for the trip of all RCPs following identification of total loss of feedwater events, operator action times for restoration of feedwater and transfer to once through cooling would be increased. Hence, the implementation of the increase in the reactor trip set point to 35.0 percent results in an overall risk reduction for total loss of feedwater transients.

One of the major inputs and assumptions used in the calculation of the NSSS design parameters is an increased minimum RCS total flow requirement to ensure that the reactor core thermal margin safety limit is not exceeded. The minimum required reactor coolant flow value with four pumps operating is determined for each refueling cycle core in the reload safety analysis and is provided in the COLR. The terminology for the trip setpoint and allowable value for reactor coolant flow-low with four reactor coolant pumps operating as well as footnote (*) for TS Table 2.2-1 is being revised to change “design reactor coolant flow” to “minimum reactor coolant flow” with a reference to COLR Table 3.2-2. The change provides a more accurate description of the actual parameter determination and ensures that the reactor coolant flow requirement is consistent with the value determined in the reload safety analysis documented in the COLR.

As discussed in LR Appendix E, the revised setpoint for the RPS reactor trip low SG level was determined using the methodology described in RIS 2006-17, NRC Staff Position on the Requirements of 10 CFR 50.36, “Technical Specifications,” Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels (Reference 6). To implement this methodology, an Allowable Value was established for the setpoint. Two new footnotes are proposed to be added to TS Table 2.2-1 for the RPS trip on low SG level providing criteria for actions related to the Allowable Value, as well as the as-found and as-left setpoint values, consistent with the recommended notes provided in the NRC letter to the NEI Technical Setpoint Methods Task Force for Setpoint Allowables (Reference 7).

8. TS 3/4.1.1.1, BORATION CONTROL – SHUTDOWN MARGIN – T_{avg} GREATER THAN 200°F

- ACTION, The minimum boron concentration is changed from “greater than or equal to 1720 ppm boron or equivalent” to “greater than or equal to 1900 ppm boron or equivalent.”

TS 3/4.1.1.2, BORATION CONTROL – SHUTDOWN MARGIN – T_{avg} LESS THAN OR EQUAL TO 200°F

- ACTION, The minimum boron concentration is changed from “greater than or equal to 1720 ppm boron or equivalent” to “greater than or equal to 1900 ppm boron or equivalent.”

Licensing Report: Section 2.1.11 Chemical and Volume Control System

Basis for the Change: A sufficient shutdown margin ensures that the reactor can be made subcritical from all operating conditions, that the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and that the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS temperature. The minimum boron concentration was increased to ensure that sufficient shutdown margin is maintained for each of the applicable design basis accidents and transients at EPU conditions. The minimum shutdown margin is calculated for each fuel cycle and the limits are provided in the COLR. The increase in the required concentration of boron in the solution is consistent with the increased boron concentration required in the refueling water tank (RWT), the safety injection tanks (SITs) and the boric acid makeup (BAM) tanks.

9. TS 3/4.1.2.2, REACTIVITY CONTROL SYSTEMS – FLOW PATHS - OPERATING

- LCO 3.1.2.2 The second set of requirements after the “OR” currently numbered “a., b., and c.” are renumbered to “d., e., and f.”

Basis for the Change: This is an administrative change that eliminates duplicate numbering of LCO requirements. The change is made for clarification and supports the change to TS 3.5.2 that references this specification.

10. TS 3/4.1.2.7, REACTIVITY CONTROL SYSTEMS – BORATED WATER SOURCES - SHUTDOWN

- LCO 3.1.2.7.a. The boric acid makeup tank parameters are changed from “a minimum borated water volume of 3550 gallons of 2.5 to 3.5 weight percent boric acid (4371 to 6119 ppm boron).” to “a minimum borated water volume of 3550 gallons of 3.1 to 3.5 weight percent boric acid (5420 to 6119 ppm boron).”
- LCO 3.1.2.7.b.2. The RWT minimum boron concentration is changed from “1720 ppm” to “1900 ppm”,

TS 3/4.1.2.8, REACTIVITY CONTROL SYSTEMS – BORATED WATER SOURCES - OPERATING

- LCO 3.1.2.8.d.2. The RWT boron concentration is changed from “between 1720 and 2100 ppm” to “between 1900 and 2200 ppm”,
- TS FIGURE 3.1-1, “ST. LUCIE 2 MIN BAMT VOLUME vs STORED BAMT CONCENTRATION” – This figure is replaced with new TS FIGURE 3.1-1, entitled “FIGURE 3.1-1, MINIMUM BAMT VOLUME vs STORED BORIC ACID CONCENTRATION”

Licensing Report: Section 2.1.11 Chemical and Volume Control System

Basis for the Change: The boron injection system ensures that negative reactivity control is available during each mode of operation. Two separate and redundant systems are provided to ensure single functional capability in the event of a failure of one of the systems. The boration capability of either system is sufficient to provide shutdown margin from all operating conditions. The EPU will have an effect on the boration requirements that must be provided by the chemical and volume control system (CVCS) boration capabilities. The EPU analysis has determined that the proposed increase in the RWT and BAM tanks minimum boron concentrations will ensure that sufficient shutdown margin is maintained at EPU conditions. The required shutdown margin is calculated for each fuel cycle and the limits are provided in the COLR.

In addition, the proposed new maximum boron concentration limit of 2200 ppm in the RWT is conservative with respect to the assumptions in the boron emergency core cooling system (ECCS) loss of coolant accident (LOCA) long-term cooling boron precipitation analysis and within the solubility limit based on the minimum temperature requirements. The proposed TS range of between 1900 and 2200 ppm in the RWT is also consistent with the assumptions in the post-LOCA containment sump pH analysis.

The revised TS Figure 3.1-1 for minimum BAM tanks volume versus boron concentration was calculated for EPU conditions and replaces the current figure.

11. TS 3/4.1.3.4 REACTIVITY CONTROL SYSTEMS – CEA DROP TIME

- LCO 3.1.3.4 is changed from requiring that the drop time of an individual CEA from a fully withdrawn reactor to its 90% insertion position be less than or equal to 3.2 seconds to less than or equal to 3.25 seconds.
- In Surveillance Requirement 4.1.3.4.b, "For specifically affected individuals CEAs…" is changed to "For specifically affected individual CEAs…"

Licensing Report: Section 2.8.5.0, Accident and Transient Analyses

Basis for Change: In order to provide additional testing margin, the safety analyses performed for the EPU have assumed a slightly longer drop time of 3.25 seconds for 90% insertion. The proposed LCO is consistent with these assumptions in the EPU safety analyses. The surveillance requirement change is editorial.

12. TS 3/4.2.5, POWER DISTRIBUTION LIMITS – DNB PARAMETERS

- LCO 3.2.5 lead-in is changed to add "of the COLR" after "Table 3.2-2",
- LCO 3.2.5.b is changed from "Pressurizer Pressure" to "Pressurizer Pressure*" to add a new footnote,
- SURVEILLANCE REQUIREMENT 4.2.5.1 is changed from "Each of the parameters of Table 3.2-2 shall be …" to "Each of the DNB-related parameters shall be …",
- SURVEILLANCE REQUIREMENT 4.2.5.2, is changed from "…within its limit by measurement* at least …" to "…within its limit by measurement** at least …" to shift the related footnote designation below the footnote that is added for LCO 3.2.5.b,
- The current footnote to TS Table 3.2-2 is added to the LCO as "* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% per minute of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.", with "per minute" added for additional clarity,
- The current footnote is revised from "*" to "**", and the threshold for the 18-month Reactor Coolant system total flow rate measurement is changed from 80% to 90%.
- Table 3.2-2 "DNB MARGIN LIMITS" is deleted from TS.

Licensing Report: Section 1.1, Nuclear Steam Supply System (NSSS) Parameters

Basis for the Change: The limits on the departure from nucleate boiling (DNB) related parameters ensure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the DNBR limit for each DNB limited transient analyzed.

The NSSS design parameters provide the RCS and secondary side system conditions (thermal power, temperatures, pressures, and flows) that are used as the basis for the design transients, systems, structures, components, accidents, and fuel analyses and evaluation. These variables are validated every refueling cycle and are contained in the COLR to provide operating and analysis flexibility.

Adding the phrase “of the COLR” to the LCO is editorial in that Table 3.2-2 is currently in the COLR and, as discussed below, is proposed to be deleted from the TS.

Adding the footnote to “Pressurizer Pressure” is administrative in that it reflects what is currently in TS Table 3.2-2 as well as in the COLR.

TS Table 3.2-2 is proposed to be deleted. It currently refers to the COLR for the identification of all DNB margin limits with the exception of “Reactor Coolant Flow Rate” which it states must be greater than or equal to the limit specified in the COLR as well as greater than or equal to 335,000 gpm. The 335,000 gpm limit was originally included as part of TS Amendment No. 138 (Reference 12) that restricted the maximum SG tube plugging level to 30 percent for the existing SGs. Because the SGs have since been replaced, the 335,000 gpm limit is no longer applicable. See Attachment 8, COLR Markups.

The changes to TS SR 4.2.5.1 are conforming changes resulting from the proposed deletion of Table 3.2-2.

TS Amendment No. 145 (Reference 13) restricted the maximum power level to 89% of 2700 MWt under certain conditions of reactor coolant flow and SG tube plugging until the Combustion Engineering Model 3410 SGs were replaced. As a conforming change, the footnote to TS SR 4.2.5.2 for RCS flow rate determination was changed from 90% RTP to 80% RTP since the power level restriction of 89% would have made flow measurement at or above 90% infeasible. The SGs have since been replaced and these restrictions are no longer applicable. Since the current maximum power level is 100% RTP, the RCS flow rate determination power level is proposed to be reverted back to 90% RTP, consistent with the earlier TS.

13. TS 3/4.4.2.2, REACTOR COOLANT SYSTEM – OPERATING

- LCO 3.4.2.2. The pressurizer code safety valve lift settings for OPERABILITY are changed from 2435.3 psig and 2535.3 psig to 2410.3 psig and 2560.3 psig.

Licensing Report: Section 2.8.4.2, Overpressure Protection During Power Operation, and Section 2.8.5.2.1 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum, and Section 2.8.5.2.4, Feedwater System Pipe Breaks Inside and Outside Containment

Basis for the Change:

The limiting transients for primary system pressure are the loss of condenser vacuum (LOCV) defined as a complete loss of steam load or a turbine trip from full power without a direct reactor trip and the feedwater line break (FWLB). The acceptance criteria for the LOCV and FWLB are 110% and 120% of RCS design pressure, respectively. Safety analyses performed under EPU conditions demonstrate that the primary safety valves in conjunction with the secondary safety valves and reactor protection system maintain the RCS below

110% of design pressures for these transients. The EPU analyses for LOCV and FWLB have been shown to support an increased tolerance of +/-3% on the as-found lift setting of the pressurizer code safety valves. The TS is therefore being changed from +/-2% to +/-3%. The as-left lift setting requirement contained in TS Surveillance 4.4.2.2 remains at +/-1%.

14. TS 3/4.4.8, REACTOR COOLANT SYSTEM – SPECIFIC ACTIVITY

- LCO 3.4.8.b, is changed from “Less than or equal to $100/\bar{E}$ microcuries/gram.” to “Less than or equal to 518.9 microcuries/gram DOSE EQUIVALENT XE-133.”,
- APPLICABILITY is changed from “1, 2, 3, 4, and 5” to “1, 2, 3, and 4”,
- ACTION: All ACTIONS, associated APPLICABILITY statements, footnote, and TS Figure 3.4-1 are deleted and replaced with the following ACTIONS:
  - a With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is 60.0 µCi/gram once per four hours.
  - b With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131, but 60.0 µCi/gram DOSE EQUIVALENT I-131, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 1.0 µCi/gram limit. Specification 3.0.4 is not applicable.
  - c With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 for greater than 48 hours during one continuous time interval, or > 60.0 µCi/gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.
  - d With the specific activity of the primary coolant > 518.9 µCi/gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 518.9 µCi/gram DOSE EQUIVALENT XE-133 limit. Specification 3.0.4 is not applicable.
  - e With the specific activity of the primary coolant > 518.9 µCi/gram DOSE EQUIVALENT XE-133 for greater than 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.
- TABLE 4.4-4, PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM, is changed as follows:
  - The “SAMPLE AND ANALYSIS FREQUENCY” column title is changed to “MINIMUM FREQUENCY.”
- Table 4.4-1, Item 1 is changed as follows:
  - TYPE OF MEASUREMENT AND ANALYSIS, “1. Gross Activity Determination” is changed to “1. DOSE EQUIVALENT XE-133 Determination.”
  - MINIMUM FREQUENCY for DOSE EQUIVALENT XE-133 is changed from “At least once per 72 hours” to “1 per 7 days.”
- TABLE 4.4-4, Item 3 and its associated footnote * are deleted, and

- TABLE 4.4-4, Item 4 is changed as follows:
  - Item “4” is changed to Item “3”,
  - TYPE OF MEASUREMENT AND ANALYSIS is changed from “…Including I-131, I-133, and I-135” to “…Including I-131, I-132, I-133, I-134, and I-135.”
  - In Item a) of MINIMUM FREQUENCY, “or 100/$\bar{E}$ micro-Ci/gram” is deleted.
  - MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED is changed from “1#, 2#, 3#, 4#, 5#” to “1#, 2#, 3# and 4#”.
- Figure 3.4-1, DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 µCi/gram Dose Equivalent I-131, is deleted.

Licensing Report: Section 2.9.2, Radiological Consequences for Alternative Source Term

Basis for the Change: The change from $\bar{E}$ to Dose Equivalent XE-133 supports the EPU analyses determination of dose consequences. TS LCO 3.4.8.b. replaces 100/$\bar{E}$ µCi/gram with 518.9 µCi/gram Dose Equivalent XE-133. This limit is established based on the RCS activity corresponding to 1 percent fuel clad defects with sufficient margin to accommodate the exclusion of some noble gas isotopes from the definition of Dose Equivalent Xe-133 based on low concentration, short half-life, or small dose conversion factors. The limit is consistent with that assumed in the accident dose consequences analyses. The primary purpose of LCO 3.4.8, RCS Specific Activity, and its associated Actions is to support the dose analyses for design basis accidents. The whole body dose is primarily dependent on the noble gas activity, not the non-gaseous activity currently included in the $\bar{E}$ definition.

TS Applicability removes Mode 5. It is necessary for the LCO to apply during Modes 1, 2, 3, and 4 to limit the potential radiological consequences of an accident that may occur during these Modes. In Mode 5, the probability of a design basis accident involving release of significant quantities of RCS inventory is greatly reduced. Therefore, the monitoring of RCS specific activity is not required. The change to modify the Applicability to Modes 1, 2, 3, and 4 retains the necessary constraints to limit the potential radiological consequences of an accident that may occur during these Modes, and is therefore acceptable from a radiological dose perspective. The MODE requirements identified for the ACTIONS are deleted to be consistent with this change in TS Applicability.

The current Actions and associated Applicability statements are deleted and replaced with new Actions a. through e. The new Actions are incorporated to implement the new LCO for Dose Equivalent XE-133 and to replace Figure 3.4-1 with a specific limit for Dose Equivalent I-131 of 60 µCi/gram. Actions a. through c. provide the requirements for Dose Equivalent I-131. Actions d. and e. provide the requirements for Dose Equivalent XE-133. The required actions and completion times are consistent with the required actions and completion times in the current TS. However, Actions c. and e. have been revised to include a requirement to be in COLD SHUTDOWN within the next 30 hours. This requirement will ensure that the unit is placed in an Operating Mode where the TS is not applicable. These changes are consistent with TSTF-490 (Reference 4).

Actions b. and d. also state, “LCO 3.0.4 is not applicable.” This will allow entry into a Mode or other specified condition in the LCO Applicability when LCO 3.4.8 is not being met. The proposed change to Action b. would allow entry into the applicable Modes from Mode 4 (HOT SHUTDOWN) through Mode 1 (POWER OPERATION) while Dose Equivalent I-131 is > 1.0 µCi/gram and 60 µCi/gram and Dose Equivalent I-131 is being restored to within its limit. The proposed change to Action d. would allow entry into the applicable Modes from Mode 4 (HOT SHUTDOWN) through Mode 1 (POWER OPERATION) while Dose Equivalent XE-133 is > 518.9 µCi/gram and Dose Equivalent XE-133 is being restored to within its limit. This Mode change is acceptable due to the significant conservatism incorporated into the Dose Equivalent I-131 and Dose Equivalent XE-133 specific activity limits, the low probability of an event occurring which is limiting due to exceeding the specific activity limits, and the ability to restore transient specific excursions while the plant remains at, or proceeds to power operation.

The middle column of TABLE 4.4-4 is changed from “SAMPLE AND ANALYSIS FREQUENCY” to “MINIMUM FREQUENCY.” TABLE 4.4-4, Primary Coolant Specific Activity Sample and Analysis Program, Type of Measurement and Analysis, Item 1 is changed from “Gross Activity Determination” to “DOSE EQUIVALENT XE-133 Determination” and the Minimum Frequency is changed from “At least once per 72 hours” to “1 per 7 days.” The change in the title of the column makes “At least…” redundant. This surveillance requires a gamma isotopic analysis as a measure of the noble gas activity of the reactor coolant at least once every seven days. The measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. The surveillance provides an indication of any increase in noble gas specific activity. Trending the results of this surveillance allows proper remedial action to be taken prior to exceeding the LCO limit under normal conditions. The surveillance frequency of 7 days considers the low probability of a gross fuel failure during this time. Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the surveillance calculation. If a specific noble gas nuclide listed in the definition for Dose Equivalent XE-133 is not detected, it will be assumed to be present at the minimum detectable activity.

TABLE 4.4-4,Item 3, “Radiochemical for $\bar{E}$ Determination” is deleted. This is consistent with the revised LCO which replaced 100/$\bar{E}$ with Dose Equivalent XE-133. Correspondingly, the footnote “*” is deleted as it is no longer applicable.

TABLE 4.4-4, Type of Measurement and Analysis, Item 4 is renumbered as Item 3 and is revised to include I-132 and I-134. This is consistent with TS 1.10 Definition of Dose Equivalent I-131. Reference to 100/$\bar{E}$ micro-Ci/gram in the Minimum Frequency column as an activity on which to base sampling is deleted for the reasons discussed above. Mode 5 is deleted as a mode in which sampling and analysis is required also for reasons discussed above.

FIGURE 3.4-1, “DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 µCi/gram Dose Equivalent I-131,” is deleted. The new limit is 60 µCi/gram Dose Equivalent I-131 at all power levels. The change from a graph that is based on the power level to a specific limit for all power levels is consistent with the dose consequence events that are analyzed at full-power and assume a pre-accident spike of 60 μCi/gram Dose Equivalent I-131. The full-power transients that allow a Dose Equivalent I-131 spike are unchanged.

15. TS 3/4.4.9.1, REACTOR COOLANT SYSTEM – PRESSURE/TEMPERATURE LIMITS

Figure 3.4-2, "ST.LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 55 EFPY, HEATUP, CORE CRITICAL, AND INSERVICE TEST" and Figure 3.4-3, "ST.LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 55 EFPY, COOLDOWN AND INSERVICE TEST" are changed to indicate that they are applicable for 47 EFPY vice 55 EFPY

Licensing Report: Section 2.1.2, Pressure–Temperature Limits and Upper Shelf Energy

Basis for the Change: The effect of changes in neutron fluence as a result of EPU conditions was evaluated to determine the impact on reactor vessel integrity. The fluence projections were prepared using the guidance of NRC Regulatory Guide (RG) 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (Reference 8). The reactor pressure vessel beltline pressure-temperature (P-T) limits are based upon irradiation damage prediction methods of NRC RG 1.99 Revision 2, Radiation Embrittlement of Reactor Vessel Materials (Reference 9). The P-T limit curves in this TS are developed in accordance with 10 CFR 50 Appendix G. The EPU evaluation projects that all reactor vessel beltline materials will continue to have an upper shelf energy (USE) value greater than 50 ft-lbs through 55 effective full power years (EFPY) which is the time corresponding to the 60 year licensed life. The acceptance criterion for USE of Appendix G is therefore met for this period. The period of applicability of the P-T limit curves, however, is based on the projections of the adjusted reference temperature (ART). The ART values that form the basis for the pre-EPU P-T limits are projected to be lower than the ART values projected under EPU conditions at 55 EFPY necessitating a change in the TS P-T limit curves. Since the current P-T limit curves will bound operation under EPU conditions if limited to 47 EFPY, the applicability of the curves is changed from 55 to 47 EFPY.

16. TS 3/4.4.9.3, REACTOR COOLANT SYSTEM – OVERPRESSURE PROTECTION SYSTEMS

- Table 3.4-3, "LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE" and Table 3.4-4, "MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP", Operating Period EFPY, is changed from 55 to 47 EFPY.
- Table 3.4-3, "Cold Leg Temperature, F°" is changed to "Cold Leg Temperature, °F"
- Table 3.4-4, the second and third columns are changed to the same format as these columns in Table 3.4-3

Licensing Report: Section 2.8.4.3, Overpressure Protection During Low Temperature Operation.

Basis for the Change: The overpressure protection system is designed to prevent violation of the RCS P-T limits in the event of an overpressure event during low temperature operation.

The design basis mass addition transients during low temperature operation are based on the operability requirements for the power operated relief valves (PORVs), the charging pumps, and the high pressure safety injection (HPSI) pumps. The maximum PORV operating pressure is determined using the nominal opening setpoint for LTOP, PORV actuation loop uncertainty, and pressure accumulation during PORV opening time. The nominal PORV setpoint is the TS value.

Peak transient pressures determined for a spectrum of cases at EPU conditions in the mass and energy addition analyses were compared to the RCS P-T limits.

The results of the analyses demonstrate that the current PORVs and shutdown cooling relief valves lift settings, and overpressure protection range can be maintained under EPU conditions without violating the existing RCS P-T limits. As described above, however, these limits will remain valid for 47 EFPY vice 55 EFPY necessitating the change to Tables 3.4-3 and 3.4-4.

The remaining changes in Tables 3.4-3 and 3.4-4 are editorial.

17. TS 3/4.5.1, EMERGENCY CORE COOLING SYSTEMS (ECCS) – SAFETY INJECTION TANKS (SIT)

- LCO 3.5.1.c. The SIT boron concentration is changed from “between 1720 and 2100 ppm of boron” to “between 1900 and 2200 ppm of boron”
- The footnote “*” in the APPLICABILITY section for Mode 3 with respect to the boron concentration requirement in the safety injection tanks, when pressurizer pressure is less than1750 psia, is also changed from “between 1720 and 2100 ppm of boron” to “between 1900 and 2200 ppm of boron”

<u>Licensing Report:</u> Section 2.8.5.6.3 Emergency Core Cooling System and Loss-of-Coolant Accidents

<u>Basis for the Change:</u> The SITs provide a sufficient volume of borated water to be immediately forced into the reactor core through each of the cold legs in the event that the RCS pressure decreases to less than the SIT pressure. The initial surge of water into the reactor core provides the initial cooling mechanism during large RCS pipe ruptures.

The SIT minimum boron concentration is increased from 1720 ppm to 1900 ppm to ensure that the core becomes subcritical and remains subcritical following a LOCA. The revised minimum boron concentration is consistent with the assumptions used for SIT injection in the safety analyses at EPU conditions. The safety analysis for post-LOCA criticality, which includes the assumption of a minimum boron concentration of 1800 ppm in the SITs, demonstrates that the containment sump boron concentration remains greater than the post-LOCA critical boron concentration to the time of ECCS switchover to hot/cold leg recirculation. The proposed TS limit of 1900 ppm is based on this minimum plus 100 ppm uncertainty. The proposed new maximum boron concentration limit of 2200 ppm is conservative with respect to the assumptions in the boron ECCS LOCA long-term cooling boron precipitation analysis and within the solubility limit based on the minimum temperature requirements. The proposed TS required boron concentration range of between 1900 and

2200 ppm in the SIT is also consistent with the assumptions in the post-LOCA containment sump pH analysis.

18. TS 3/4.5.2, ECCS SUBSYSTEMS – OPERATING

- LCO 3.5.2.d, a new Footnote, "*" is added stating "* One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e."
- APPLICABILITY is changed from "MODES 1, 2 and 3*" to "MODES 1, 2 and 3**".
- Footnote "* With pressurizer pressure greater than or equal to 1750 psia." is changed to "** With pressurizer pressure greater than or equal to 1750 psia."
- Surveillance Requirement 4.5.2.f.1. is revised from "… each automatic valve in the flow path actuates …" to "… each automatic valve in the flow paths actuates …",
- Surveillance Requirement 4.5.2.f.2.a. is revised from "High-Pressure Safety Injection pump." to "High-Pressure Safety Injection pumps.",
- Surveillance Requirement 4.5.2.f.2.b. is revised from "Low-Pressure Safety Injection pump." to "Low-Pressure Safety Injection pumps.",
- A new Surveillance Requirement 4.5.2.f.2.c. "Charging Pumps." is added,
- Surveillance Requirement 4.5.2.g. an underline "_" is deleted.

<u>Licensing Report:</u> Section 2.8.5.6.3, Emergency Core Cooling System and Loss-of-Coolant Accidents

<u>Basis for the Change:</u> The operability of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming loss of one subsystem from a single failure. In addition, each ECCS subsystem provides long-term cooling capability in the recirculation mode during the accident recovery period.

The footnote to LCO 3.5.2.d, "One OPERABLE charging pump" provides a cross reference to TS 3.1.2.2, REACTIVITY CONTROL SYSTEMS – FLOW PATHS - OPERATING, to alert the operator that inoperability of one or both charging pumps may impact TS 3.1.2.2. Minor corrections are made to the safety injection signal testing surveillances to assure that it is clear that the flow paths rather than a single flow path is tested, and that both of the HPSI pumps and low pressure safety injection (LPSI) pumps are tested. In addition, the charging pumps are added to surveillance requirements for safety injection signal testing and are included in the Inservice Testing (IST) Program.

These changes are not EPU required changes, but are valid enhancements that assure ECCS sources credited in the analysis are available.

19. TS 3/4.5.4, EMERGENCY CORE COOLING SYSTEMS – REFUELING WATER TANK

- LCO 3.5.4.b. The RWT boron concentration is changed from "between 1720 and 2100 ppm" to "between 1900 and 2200 ppm."

Licensing Report: Section 2.8.5.6.3, Emergency Core Cooling System and Loss-of-Coolant Accidents

Basis for the Change: The RWT, as part of the ECCS, ensures that a sufficient supply of borated water is available for injection into the reactor core in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that sufficient water is available within the containment to permit recirculation cooling flow to the core, and that the reactor remains subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control element assemblies (CEA) inserted, except for the most reactive CEA. The RWT minimum boron concentration is reviewed for each cycle-specific core design to confirm that adequate boron exists to maintain subcriticality in a long-term post-LOCA environment.

The safety analysis for post-LOCA criticality, which includes the assumption of a minimum boron concentration of 1800 ppm in the RWT, demonstrates that the containment sump boron concentration remains greater than the post-LOCA critical boron concentration to the time of ECCS switchover to hot/cold leg recirculation. The TS limit is based on this minimum plus 100 ppm uncertainty.

The proposed new maximum boron concentration limit of 2200 ppm is conservative with respect to the assumptions in the boron ECCS LOCA long-term cooling boron precipitation analysis and within the solubility limit based on the minimum temperature requirements. The proposed TS required boron concentration range of between 1900 and 2200 ppm in the RWT is also consistent with the assumptions in the post-LOCA containment sump pH analysis.

20. TS 3/4.7.1.1 PLANT SYSTEMS – TURBINE CYCLE – SAFETY VALVES

- Table 3.7-2 “STEAM LINE SAFETY VALVES PER LOOP”
  - Column header, “LIFT SETTING (+ 1% to - 3%)” is changed to “LIFT SETTING *”,
  - The upper limit for valves 8201, 8202, 8203, 8204, 8205, 8206, 8207, and 8208 is changed from “995.3 psig” to “1015.3 psig”, and
  - The upper limit for valves 8209, 8210, 8211, 8212, 8213, 8214, 8215, and 8216 is changed from “1035.7 psig” to “1046.1 psig”.
  - A new Footnote is added to read, “* +/-3% for valves a through d and +2%/-3% for valves e through h”.

Licensing Report: Section 2.5.5.1 Main Steam and Section 2.8.5.2.1 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum

Basis for the Change: This is not an EPU related change. The current main steam safety valve (MSSV) nominal setpoints, 1000 psia and 1040 psia, remain unchanged for the EPU. Current MSSV lift setpoints, shown in TS Table 3.7-2, are based on lift tolerances of +1/-3% for all valves. The positive setpoint tolerance for both banks of MSSVs is being revised to provide operating margin and allow for setpoint drift. The revised setpoint tolerances are:

- For MSSVs with a nominal setpoint of 1000 psia, the positive setpoint tolerance is being changed to +3%.

- For MSSVs with a nominal setpoint of 1040 psia, the positive setpoint tolerance is being changed to +2%.

The revised safety analysis supports these limits by modeling the revised tolerance for each of the banks of MSSVs.

21. TS 3/4.8.1, ELECTRICAL POWER SYSTEMS – A.C. SOURCES

TS 3/4.8.1.1 A.C. SOURCES – OPERATING

- LCO 3.8.1.1.b.2. is being changed from: “A separate fuel storage system containing a minimum volume of 40,000 gallons of fuel,” to: “A separate fuel storage system containing a minimum volume of 42,500 gallons of fuel,”
- Surveillance Requirement 4.8.1.1.2.e.4.b. The voltage is changed from “…4160 +/- 420…” to “…4160 +/- 210…”, and the frequency is changed from “…60 +/- 1.2 Hz…” to “… 60 +/- 0.6 Hz…”,
- Surveillance Requirement 4.8.1.1.2.e.5 is rewritten as indicated below:

  5 Verifying that on an ESF actuation test signal (without loss-of-offsite power) the diesel generator starts**** on the auto-start signal, and:

    a Within 10 seconds, generator voltage and frequency shall be 4160 +/- 420 volts and 60 +/- 1.2 Hz.

    b Operates on standby for greater than or equal to 5 minutes.

    c Steady-state generator voltage and frequency shall be 4160 +/- 210 volts and 60 +/- 0.6 Hz and shall be maintained throughout this test.,

- Surveillance Requirement 4.8.1.1.2.e.6.b): The voltage is changed from “…4160 +/- 420…” to “…4160 +/- 210…”, and the frequency is changed from “…60 +/- 1.2 Hz…” to “… 60 +/- 0.6 Hz…”, and
- Surveillance Requirement 4.8.1.1.2.e.7 is rewritten as indicated below:

  7 Verifying that the diesel operates for at least 24 hours****.

    a Within 10 seconds, generator voltage and frequency shall be 4160 +/- 420 volts and 60 +/- 1.2 Hz.

    b Steady-state generator voltage and frequency shall be 4160 +/- 210 volts and 60 +/- 0.6 Hz and shall be maintained throughout this test.

    c During the first 2 hours of this test, the diesel generator shall be loaded within a load band of 3800 to 3985 kW#, and

    d During the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3450 to 3685 kW#.

TS 3/4.8.1.2 A.C. SOURCES – SHUTDOWN

- LCO 3.8.1.2.b.2. is being changed from: “A fuel storage system containing a minimum volume of 40,000 gallons of fuel, and” to: “A fuel storage system containing a minimum volume of 42,500 gallons of fuel, and”

Licensing Report: Section 2.5.7.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System and Section 2.2.4 Safety-Related Valves and Pumps

Basis for the Change: While the current TS requires a minimum of 40,000 gallons of fuel in each diesel oil storage tank, administrative controls require 42,500 gallons. The administrative requirement was imposed when ultra low sulfur diesel (ULSD) fuel replaced low sulfur diesel fuel to meet the requirements of the Clean Air Act. ULSD may have a lower heat rate and, therefore, necessitated an increase in the storage requirement. An analysis of the diesel oil storage tank requirements under EPU conditions has determined that, although peak loading of the emergency diesel generators (EDGs) has increased, the total integrated load on an EDG over the course of seven days after a LOCA coincident with a loss of offsite power is such that the current administrative limit of 42,500 gallons is still sufficient including 10% margin. The proposed change to the TS for the minimum required volume, therefore, is based on the incorporation of the increased volume requirements resulting from the use of ULSD.

The proposed revisions to the voltage and frequency tolerances for the EDG are not required by the EPU. The need to revise the EDG frequency and voltage was identified and entered into the corrective action program as condition report 2007-23473 entitled Impact on EDG Frequency Tolerance on Plant Equipment. This proposed change is a corrective action necessary to close the condition report. In addition, NRC Inspection Report No. 05000335/2007006 and 05000389/2007006 for a component design bases inspection conducted in September 2007, documented that the EDG voltage and frequency tolerance issue had been self-identified and was previously entered into the site corrective action program. As such, the NRC finding was determined to be of very low significance and treated as a non-cited violation. However, the EDG and the components loaded onto the EDG were evaluated under EPU conditions for any mechanical or electrical impact from the revised tolerances for steady state frequency, 60 +/- 0.6 Hz (+/- 1%), and steady state voltage, 4160 +/- 210V (+/- 5%). These tolerances reflect worst case values used in determining motor-operated valve (MOV) and pump loads connected to an EDG and are considered in the accident analyses.

The change in frequency tolerance has been evaluated for changes in MOV stroke times and pump flow rates. The IST program acceptance criteria will be verified for MOVs and pump flows during the EPU Implementation Phase.

The change in voltage tolerance of +/- 5% will not affect the motor speed of MOVs with AC motors powered from the EDGs, and therefore the stroke time of these MOVs will not be affected. Evaluation shows that the voltage tolerance of +/- 5% will not affect the minimum motor terminal voltage values used in determining MOV motor torque values under degraded voltage conditions.

The proposed tightening of the TS steady state frequency and voltage tolerances is conservative, and an enhancement to the TS. The changes associated with voltage and frequency tolerances are not required for EPU.

22. TS 3/4.9.1, REFUELING OPERATIONS – BORON CONCENTRATION

- ACTION: The ACTION requirement is changed from "…continue boration at greater than or equal to 40 gpm of a solution containing 1720 ppm boron or greater…" to "…continue boration at greater than or equal to 40 gpm of a solution containing 1900 ppm boron or greater…"

Licensing Report: Section 2.1.11 Chemical and Volume Control System

Basis for the Change: The limitation on minimum boron concentration ensures that the reactor will remain subcritical during core alterations, and a uniform boron concentration is maintained for reactivity control in the water volumes having direct access to the reactor vessel. The limitation on K_{eff} is sufficient to prevent reactor criticality with all full length rods (shutdown and regulating) fully withdrawn. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving core alterations are suspended immediately.

The revised minimum concentration of 1900 ppm for boration flow flow to the RCS and refueling cavity ensures that the core will remain subcritical during refueling operations involving core alterations or positive reactivity changes under EPU conditions.

23. TS 3/4.9.11, REFUELING OPERATIONS – SPENT FUEL STORAGE POOL

- LCO 3.9.11.b. is changed from "greater than or equal to 1720 ppm." to "greater than or equal to 1900 ppm."

Licensing Report: 2.8.6.2 Spent Fuel Storage

Basis for the Change: The spent fuel storage pool minimum boron concentration is increased from 1720 ppm to 1900 ppm to ensure that the core will remain subcritical whenever irradiated fuel assemblies are in the spent fuel storage pool under EPU conditions. The limit on soluble boron concentration is consistent with the minimum boron concentration specified for the RWT, and assures an additional subcritical margin to the value of K_{eff} which is calculated in the spent fuel storage pool criticality safety analysis. Inadvertent dilution of the spent fuel storage pool by a quantity of unborated water necessary to reduce the pool boron concentration to a value that would invalidate the criticality safety analysis is not considered a credible event.

24. TS 3/4.10.1, SPECIAL TEST EXCEPTIONS – SHUTDOWN MARGIN

- ACTION a. is changed from "…continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent" to "…continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or its equivalent", and
- ACTION b. is changed from "…continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to1720 ppm boron" to "…continue boration at

greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron."

Licensing Report: Section 2.1.11 Chemical and Volume Control System

Basis for the Change: This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEA worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations. The revised minimum concentration of 1900 ppm for the boration flow to the RCS is consistent with the requirements for the RWT, and will provide sufficient shutdown margin for EPU conditions if the reactivity equivalent of the highest estimated CEA is not available for trip, with any full-length CEA fully withdrawn, or if the reactor is subcritical by less than the shutdown margin of TS 3.1.1.1 with all full length CEAs withdrawn.

25. TS 3/4.11.2.6, RADIOACTIVE EFFLUENTS – GAS STORAGE TANKS

- LCO 3.11.2.6 is changed from "…less than or equal to 285,000 curies noble gases…" to "…less than or equal to 202,500 curies noble gases…", and
- Surveillance Requirement 4.11.2.6 is changed from "…when reactor coolant system activity exceeds $100/\bar{E}$" to "…when reactor coolant system activity exceeds 518.9 μCi/gram DOSE EQUIVALENT XE-133."

Licensing Report: Section 2.9.3 Radiological Consequences of Gas Decay Tank Rupture and Section 2.9.2, Radiological Consequences for Alternative Source Term.

Basis for the Change: The waste gas decay tank inventory source term required to generate an exclusion area boundary (EAB) dose of 0.1 rem total effective dose equivalent (TEDE) is the basis for a proposed TS limit of 202,500 Dose Equivalent Curies Xe-133. The limit of 0.1 rem is consistent with Branch Technical Position (BTP) 11-5, Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure, of Standard Review Plan (SRP) Chapter 11, Radioactive Waste Management, of NUREG-0800 (Reference 11). The RCS specific activity value of 518.9 μCi/gram for Dose Equivalent XE-133 is discussed in Change No. 13, TS 3/4.4.8, REACTOR COOLANT SYSTEM – SPECIFIC ACTIVITY, above.

Waste gas decay tank source term has been recalculated for the EPU. The waste gas decay tank inventory source term required to generate an exclusion area boundary dose close of 0.1 rem TEDE is the basis for a proposed TS limit of 202,500 Dose Equivalent Curies Xe-133. The RCS specific activity value of 518.9 μCi/gram for Dose Equivalent XE-133 is discussed in Change No.14, TS 3/4.4.8, REACTOR COOLANT SYSTEM – SPECIFIC ACTIVITY, above.

26. TS 5.6, DESIGN FEATURES – FUEL STORAGE – CRITICALITY

- TS 5.6.1.a.2 is changed from "…containing 520 ppm boron,…" to "…containing 500 ppm boron,…"
- TS 5.6.1.a.3 is changed from "A nominal 8.96 inch center-to-center distance between fuel assemblies…" to "A nominal 8.965 inch center-to-center distance between fuel assemblies…"

- TS 5.6.1.a.4 is re-designated as TS 5.6.1.b
- New TS 5.6.1.a.4 and 5.6.1.a.5 are added as follows:

  4 For storage of enriched fuel assemblies, requirements of Specification 5.6.1.a.1 and 5.6.1.a.2 shall be met by positioning fuel in the spent fuel pool storage racks consistent with the requirements of Specification 5.6.1.c or in configurations that have been shown to comply with Specifications 5.6.1.a.1 and 5.6.1.a.2 using the methodology as described in Section 9.1 of the Updated Final Safety Analysis Report.

  5 Fissile material, not contained in a fuel assembly lattice, shall be stored in accordance with the requirements of Specifications 5.6.1.a.1 and 5.6.1.a.2.

- TS 5.6.1.b.1 through 5.6.1.b.4 are replaced with the following:

  b The cask pit storage rack shall contain neutron absorbing material (Boral) between stored fuel assemblies when installed in the spent fuel pool.

- TS 5.6.1.c.1 through 5.6.1.c.3 are replaced with the following:

  c Loading of spent fuel pool storage racks shall be controlled as described below.

    1 The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel pool storage rack shall be less than or equal to 4.6 weight percent.

    2 Fuel placed in Region 1 of the spent fuel pool storage racks shall comply with the storage pattern definitions of Figure 5.6-1 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)

    3 Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions or allowed special arrangement definitions of Figure 5.6-2 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)

    4 The 2x2 array of fuel assemblies that span the interface between Region 1 and Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions of Figure 5.6-3 and the minimum burnup requirements as defined in Table 5.6-1. The allowed special arrangements in Region 2 as shown in Figure 5.6-2 shall not be placed adjacent to Region 1. (See Specification 5.6.1.c.7 for exceptions)

    5 Fuel placed in the cask pit storage rack shall comply with the storage pattern definitions of Figure 5.6-4 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)

    6 The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required.

    7 Fresh or spent fuel in any allowed configuration may be replaced with non-fuel hardware, and fresh fuel in any allowed configuration may be replaced with a fuel rod storage basket containing fuel rod(s). Also, storage of Metamic inserts or control rods, without any fissile material, is acceptable in locations designated as completely water-filled cells.

- TS 5.6.1.d is changed from "…having a U-235 enrichment less than or equal 4.5 weight percent…" to "…having a maximum planar average U-235 enrichment less than or equal 4.6 weight percent…"
- TS 5.6.3 is changed from "… with a storage capacity limited to not more than 1360 fuel assemblies…spent fuel pool and cask pit storage capacity is limited to no more than 1585 fuel assemblies" to "…with a storage capacity limited to not more than 1491 fuel assemblies…spent fuel pool and cask pit storage capacity is limited to no more than 1716 fuel assemblies"
- Figure 5.6-1a on page 5-4b is deleted and replaced on the page with the following note:
- "Pages 5-4C through 5-4F (Amendment 101) and page 5-4G (Amendment 135) have been deleted from the Technical Specifications. The next page is 5-4h."
- Pages 5-4C through 5-4G (with Figures 5.6-1b through 5.6-1f) are deleted.
- The following new figures are added:
  Figure 5.6-1 – Allowable Region 1 Storage Patterns and Fuel Arrangements
  Figure 5.6-2 (Sheets 1, 2 and 3) - Allowable Region 2 Storage Patterns and Fuel Arrangements
  Figure 5.6-3 (Sheets 1 and 2) – Interface Requirements between Region 1 and Region 2
  Figure 5.6-4 – Allowable Cask Pit Storage Rack Patterns
- Table 5.6-1 – Minimum Burnup Coefficients is added.

<u>Licensing Report:</u> Sections 2.8.2 Nuclear Design; 2.8.6.1 New Fuel Storage; 2.8.6.2 Spent Fuel Storage; and Appendix G to Attachment 5, St. Lucie Unit 2 Criticality Analysis for EPU and Non-EPU.

<u>Basis for the Change:</u> Changes to Design Features - Fuel Storage - Criticality are based on the analysis provided in the Appendix G to Attachment 5 (criticality analysis). It replaces the CLB analysis with an analysis that bounds both pre-EPU and EPU fresh and spent fuel with an initial enrichment of 1.5 to 4.6 weight percent U-235. The spent fuel racks and cask pit racks analyses ensure that k_{eff} is less than or equal to 0.95 with the storage racks fully loaded with fuel of the highest anticipated reactivity and the pool flooded with borated water at a temperature corresponding to the highest reactivity. In addition, the analysis demonstrates that k_{eff} is less than 1.0 with the storage racks fully loaded with fuel of the highest anticipated reactivity and the pool flooded with unborated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivities include a margin for uncertainty, including manufacturing tolerances, and are calculated with a 95% probability at a 95% confidence level. Reactivity effects of abnormal and accident conditions in the cask pit storage rack and spent fuel racks have also been evaluated to assure that under all credible abnormal and accident conditions, the reactivity will not exceed 0.95 with credit for soluble boron.

The changes to TS 5.6.1.a.2 and 5.6.1.a.3 are made to conform to the assumptions in the criticality analysis.

Current TS 5.6.1.a.4 is re-designated as TS 5.6.1.b.

The revised TS 5.6.1.a.4 requires that the storage of enriched fuel assemblies meet the requirements in TS 5.6.1.a.1 and TS 5.6.1.a.2 by storage consistent with TS 5.6.1.c or in configurations shown to comply with these requirements using the methodology in the UFSAR. Storage in this manner is consistent with the criticality analysis.

New TS 5.6.1.a.5 ensures that fissile material not in a fuel assembly lattice meets the regulatory criteria required by the referenced TS 5.6.1.a.1 and 5.6.1.a.2 consistent with the criticality analysis.

The requirements for spent fuel pool Region 1 and Region 2, currently in TS 5.6.1.b and TS 5.6.1.c, are revised consistent with the criticality analysis and included in revised TS 5.6.1.c.

The proposed revised TSs 5.6.1.c.1 through 5.6.1.c.5 provide configurations, allowable in accordance with the criticality analysis, for the storage of EPU fuel, existing pre-EPU fuel and fresh fuel in the Region 1 and Region 2 storage racks as well as the cask pit storage rack with enrichment up to a maximum initial planar average value of 4.6 weight percent U-235. As part of revised TS 5.6.1.c, the designations for the spent fuel regions are changed from “Region I” and “Region II” to “Region 1” and “Region 2”, respectively.

The proposed TS 5.6.1.c.6 provides a requirement for directional orientation for Metamic inserts for contiguous groups of 2x2 arrays consistent with the criticality analysis.

The proposed TS 5.6.1.c.7 provides, in accordance with the criticality analysis, that insertion of non-fuel hardware in the cells allowed to contain fresh or spent fuel is permitted; that the storage of Metamic inserts or control rods is acceptable; and that fresh fuel in any configuration may be replaced with a fuel rod storage basket containing fuel rod(s).

The revised TS 5.6.1.d reflects the results of a new bounding evaluation that was performed to assess the storage of fresh fuel at a enrichment level that may be used as part of EPU. The evaluation determined that the new fuel storage racks, when filled with fresh fuel having a maximum planar average enrichment of 4.6 weight percent U-235, support EPU activities and meet regulatory requirements.

TS Figures 5.6-1a through 5.6-1f are deleted and are replaced with new Figures 5.6-1 through 5.6-4 which provide the requirements for determining EPU and pre-EPU spent fuel and fresh fuel storage configurations and arrangements in Region 1, Region 2, the interfaces between Region 1and 2, and in the cask pit storage rack that conform to the criticality analysis of Appendix G to Attachment 5.

New Table 5.6-1, Minimum Burnup Coefficients, is added. This table provides the coefficients with which to determine the minimum burnup for a fuel type and the cooling times, as applicable, which are then used in conjunction with Figures 5.6-1 through 5.6-4 to determine allowable storage configurations under the criticality analysis.

Revised TS 5.6.3 reflects the new licensing basis criticality analyses, as well as the thermal and structural analyses, which support increasing the TS limit for available spent fuel pool rack storage capacity from 1360 cells to 1491 cells.

27. TS 6.8.4, ADMINISTRATIVE CONTROLS – PROGRAMS

- Paragraph "a)" of TS 6.8.4.h, Containment Leakage Rate Testing Program, is incorporated into the introductory paragraph.
- TS 6.8.4.h.b) is deleted.
- The second paragraph of TS 6.8.4.h is changed from "The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 41.8 psig." to "The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 43.48 psig."

Licensing Report: Section 2.6.1 Containment Functional Design

Basis for the Change: TS 6.8.4.h.b) is deleted because it refers to an event in the past that is not relevant to the current TS ("The first Type A test performed after the June 1992 Type A test shall be prior to startup following the SL2-17 refueling outage.")

The peak containment pressure from the design basis LOCA is increased from the current value of 41.8 psig to 43.48 psig. This is less than the containment design pressure of 44 psig. The initial containment pressure assumed in the analysis is 15.41 psia which is consistent with the TS 3.6.1.4 maximum containment internal pressure of 0.4 psig plus 0.31 psig instrument uncertainty.

28. TS 6.9.1.11, ADMINISTRATIVE CONTROLS – CORE OPERATING LIMITS REPORT (COLR)

- TS 6.9.1.11.b: The references for the COLR are revised to address the revised analyses for EPU. The following references are deleted and replaced with "DELETED":

3 CENPD-199-P, Rev. 1-P-A, "C-E Setpoint Methodology: CE Local Power

Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986.

4 CENPD-266-P-A, "The ROCS and DIT Computer Code for Nuclear Design," April 1983.

6 CENPD-188-A, "HERMITE: A Multi-Dimensional Space – Time Kinetics Code for PWR Transients," July 1976.

7 CENPD-153-P, Rev. 1-P-A, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System," May 1980.

9 CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for St. Lucie Unit 1," January 1980.

13 CEN-371(F)-P, "Extended Statistical Combination of Uncertainties," July 1989.

15 CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.

16 CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution," April 1975.

17 CENPD-207-P-A, “Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-uniform Axial Power Distribution,” December 1984.

18 CENPD-206-P-A, “TORC Code, Verification and Simplified Modeling Methods,” June 1981.

37 Letter, A.E. Scherer Enclosure 1-P to LD-82-001, “CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System,” December 1981.

38 Safety Evaluation Report, “CESEC Digital Simulation of a Combustion Engineering Steam Supply System (TAC No.: 01142),” October 27, 1983.

39 CENPD-282-P-A, Volumes 1, 2 and 3, and Supplement 1, “Technical Manual for the CENTS Code,” February 1991, February 1991, October 1991, and June 1993, respectively.

40 CEN-121(B)-P, “CEAW, Method of Analyzing Sequential Control Element Assembly Group Withdrawal Event for Analog Protected Systems,” November 1979 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

41 CEN-133(B), “FIESTA, A One Dimensional, Two Group Space-Time Kinetics Code for Calculating PWR Scram Reactivities,” November 1979 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

44 CENPD-183-A, “C-E Methods for Loss of Flow Analysis,” June 1984.

45 CENPD-190-A, “C-E Method for Control Element Assembly Ejection Analysis,” July 1976.

46 CENPD-199-P, Rev. 1-P-A, Supplement 2-P-A, “CE Setpoint Methodology,” June 1998.

47 CENPD-382-P-A, “Methodology for Core Designs Containing Erbium Burnable Absorbers,” August 1993.

53 CEN-365(L), “Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis,” June 1988 (NRC SER dated March 13, 1989, Letter J.A. Norris (NRC) to W.F. Conway (FPL), TAC No. 69325).

54 DP-456, F.M. Stern (CE) to E. Case (NRC), dated August 19, 1974, Appendix 6B to CESSAR System 80 PSAR (NRC SER, NUREG-75/112, Docket No. STN 50-470, “NRC SER – Standard Reference System, CESSAR System 80,” December 1975).

- TS 6.9.1.11.b: Reference 64 is being rewritten to provide additional information as follows:

64 Letter, W. Jefferson Jr. (FPL) to Document Control Desk (USNRC), “St. Lucie Unit 2 Docket No. 50-389: Proposed License Amendment WCAP-9272 Reload Methodology and Implementing 30% Steam Generator Tube Plugging Limit,” L-2003-276, December 2003 (NRC SER dated January 31, 2005, Letter B.T. Moroney (NRC) to J.A. Stall (FPL), TAC No. MC1566).

- TS 6.9.1.11.b: The following references are being added:

65 WCAP-14882-P-A, Rev. 0, " RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

66 WCAP-7908-A, Rev. 0, "FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod," December 1989.

67 WCAP-7979-P-A, Rev. 0, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," January 1975.

68 WCAP-7588, Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Special Kinetics Methods," January 1975.

- TS 6.9.1.11.b: The supplement to Revision 1-P is being added to Reference 5, CENPD-275-P as follows:

5 CENPD-275-P, Revision 1-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," May 1988, & Revision 1-P Supplement 1-P-A, April 1999.

Basis for the Change: These changes incorporate the changes to the analytical methods used to determine the core operating limits. All methods have been previously approved by the NRC. References that are no longer used are deleted and new references have been included.

3.2 Licensing Basis Changes

1. Related License Amendment Requests (LARs)

There are no other EPU-related LARs pending NRC approval.

2. Dose Analyses

FPL has calculated doses under EPU conditions for all design basis accidents (DBAs) required by NRC RG 1.183 (Reference 10), and NUREG-0800, Section 15.0.1 Radiological Consequences Analyses Using Alternative Source Terms (Reference 11). Doses were calculated for the EAB, low population zone (LPZ), and control room for each accident. The dose analyses for the EPU indicate that the increase in control room dose for the LOCA, steam generator tube rupture (SGTR), locked rotor, letdown line rupture, fuel handling accident (FHA), and CEA ejection design basis accidents (DBAs) exceed the threshold for minimal increase under 10 CFR 50.59, defined as ten percent of the difference between the current calculated dose value and the regulatory guideline value. In addition, the EPU dose analysis for the locked rotor design basis accident yields an increase very slightly above this threshold for the LPZ. These doses are acceptable because they remain less than the limits established in 10 CFR 50.67 Accident Source Term, and the acceptance criteria contained in RG 1.183 and SRP 15.0.1. The DBA dose analyses and results are summarized in LR Section 2.9.2, Radiological Consequences Analyses Using Alternative Source Term.

In the case of the WGDT rupture, the pre- and post-EPU offsite doses cannot be directly compared. As described in LR 2.9.3 Radiological Consequences of Gas Decay Tank Rupture, the CLB analysis is based on meeting 10 CFR Part 100 while the EPU analysis is conducted using the RG 1.183 alternative source term and BTP 11-5 criteria. For control room doses, the limit of 5.0 rem TEDE in 10 CFR 50.67 (General Design Criterion (GDC)-19) is used in both the pre- and post-EPU dose analyses. Although the TS limit on WGDT inventory is based on limiting the radiological dose at the EAB to 0.1 rem TEDE in the event of their release, the actual calculated doses of a WGDT rupture under EPU conditions for the EAB, as well as the LPZ and control room, are well below regulatory limits. The WGDT dose analyses and results are summarized in LR Section 2.9.3, Radiological Consequences for Gas Decay Tank Ruptures.

3. Plant Modifications

   Installation of the Cameron LEFM CheckPlus™ system, as described in LR 2.4.4, Measurement Uncertainty Recapture Power Uprate, will be performed pursuant to 10 CFR 50.59. Approval of this LAR, however, is required to use the Cameron LEFM CheckPlus™ system to perform calorimetric calculations with a 0.3 percent uncertainty.

4.0 TECHNICAL ANALYSIS

LAR Attachment 5, Licensing Report summarizes the evaluations performed to assure acceptable operation at EPU conditions, and provides technical justification for the EPU related changes to the RFOL, TS and CLB, as well as the plant modifications.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

5.2 No Significant Hazards Consideration

The proposed license amendment will revise the St. Lucie Unit 2 renewed facility operating license (RFOL) NPF-16 and the technical specifications (TS) to increase licensed core thermal power from 2700 megawatts thermal (MWt) to 3020 MWt. This represents a net increase in licensed thermal power of approximately 11.85 percent and includes a 10.0 percent power uprate and a 1.7 percent measurement uncertainty recapture (MUR). The net increase is calculated as follows:

$$(2700 \text{ MWt} \times 1.10) \times 1.017 \cong 3020 \text{ MWt}$$

$$(3020 \text{ MWt} - 2700 \text{ MWt}) / 2700 \text{ MWt} \cong 11.85\%$$

Due to the magnitude of this increase in licensed thermal power, this power uprate is defined as an extended power uprate (EPU). In addition, the proposed amendment includes changes to the St. Lucie Unit 2 current licensing basis (CLB). The changes to the RFOL, TS, and CLB have been grouped and evaluated pursuant to 10 CFR 50.92.

A. Reactor Core Power Level

   The RFOL Maximum Power Level and the TS Definition for Rated Thermal Power (RTP) are changed from 2700 MWt to 3020 MWt. This represents a net increase in licensed thermal power of approximately 11.85 percent and includes a 10.0 percent power uprate and a 1.7 percent measurement uncertainty recapture.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The EPU evaluations demonstrate that the increased thermal power will continue to allow safe operation of the plant and will not affect the health and safety of the public.

   An evaluation of the components and systems, including interface and control systems that could be affected by the increase in power level, was performed for the EPU. Changes required to support the increase, including operating parameters changes, plant modifications, and procedure changes, will ensure that the components and systems will continue to meet their intended design basis functions under EPU conditions.

   The EPU evaluations included performance of accident analyses at EPU conditions. NRC-approved methodologies were used in the performance of the accident analyses. All restrictions and limitations of these methodologies have been met in the application of these methodologies to the EPU accident analyses. The results of these analyses demonstrate that applicable acceptance criteria continue to be satisfied.

   Dose consequences were evaluated using EPU parameters. The calculated doses remain less than the applicable limits established in 10 CFR 50.67, Accident Source Term, and the acceptance criteria contained in NRC Regulatory Guide (RG) 1.183, Branch Technical Position (BTP) 11-5, and General Design Criterion (GDC)-19.

   Therefore, the proposed power uprate does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No new accident scenarios are introduced as a result of the proposed changes. Modified components do not introduce any significant failures different from those of the components in the pre-modified conditions. Structures, systems and components (SSCs) previously required for transient mitigation remain capable of fulfilling their intended design functions.

   The increase in power level does not create new fission product release paths. The fission product barriers (fuel cladding, reactor coolant pressure boundary, and the containment) remain unchanged.

   Operating procedure changes do not result in any significant changes in operating philosophy. Accordingly, the proposed power uprate does not introduce human performance issues that would create new accidents or different accident sequences.

   Therefore, the proposed power uprate does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The margins of safety associated with the power uprate are those pertaining to core thermal power. These include fuel cladding, reactor coolant pressure boundary, and containment barriers. The EPU analyses demonstrated that the fuel system design and the thermal and hydraulic design are acceptable at EPU conditions. Cycle-specific analyses will be performed to confirm that fuel design limits are not exceeded. The EPU analyses demonstrated that reactor coolant pressure boundary integrity will be maintained at EPU conditions. The analyses also demonstrated that the design basis limits for the containment continue to be satisfied at EPU conditions. The accident and transient analyses demonstrated that the applicable acceptance criteria are met at EPU conditions.

   Therefore, the proposed change does not result in a significant reduction in a margin of safety.

B. Departure From Nucleate Boiling (DNB) Parameters

The TS curves providing the reactor core thermal margin safety limit lines are revised to reflect the EPU analyses and the table identifying DNB margin limits is deleted.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Safety limits such as the reactor core thermal margin safety limit lines are used to mitigate the consequences of an accident, and are not accident initiators. The TS DNB Margin Limits table currently refers to the core operating limits report (COLR) for identification of DNB margin limits and states that, for Mode 1 operation, the reactor coolant flow rate must be greater than or equal to 335,000 gpm and the COLR limit. As described above, the 335,000 gpm limit was based on a limitation in place from startup for Cycle 16 until replacement of the steam generators (SGs). The SGs have been replaced. The TS table is therefore duplicative of the COLR. The COLR variables are validated every fuel cycle.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Safety limits such as the reactor core thermal margin safety limit lines are used to mitigate the consequences of an accident, and are not accident initiators. The DNB-related parameters limits are consistent with the safety analyses assumptions and are included in the COLR and validated each fuel cycle. The DNB-related parameters are not accident initiators. Accordingly, DNB-related parameters do not create the probability of a new or different kind of accident.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The reactor core thermal margin safety limit lines reflect the EPU accident and transient analyses as appropriate. The TS DNB Margin Limits table currently refers to the COLR for identification of DNB margin limits and states that, for Mode 1 operation, the Reactor Coolant flow rate must be greater than or equal to 335,000 gpm and the COLR limit. As described above, the 335,000 gpm limit was based on a limitation in place from startup for Cycle 16 until replacement of the SGs. The SGs have been replaced. The TS table is therefore duplicative of the COLR. The COLR variables are validated every fuel cycle.

   Therefore, the proposed changes do not result in a significant reduction in a margin of safety.

C. Steam Generator Water Level Limiting Safety System Setting

The SG water level low trip setpoint is being changed from 20.5% to 35.0% and its associated allowable value is changed from 19.5% to 34.1%.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Limiting safety system settings are used to mitigate the consequences of an accident, and are not accident initiators. Accordingly, limiting safety system settings do not increase the probability of an accident previously evaluated.

   The accidents and transient analyses for EPU were conservatively performed using the current TS value for reactor trip on low SG level and the results are within acceptable limits. This TS change provides additional margin to postulated total loss of feedwater flow events. Accordingly, the increase in the reactor trip setpoint for SG water level low results in an overall risk reduction for total loss of feedwater transients.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Limiting safety system settings are used to mitigate the consequences of an accident, and are not accident initiators. Accordingly, limiting safety system settings do not create the probability of a new or different kind of an accident.

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The change has no adverse affect on any safety-related system and does not change the performance or integrity of any safety-related system. Additionally, no new safety-related equipment is being added or replaced as a result of the proposed change to the trip setpoint.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The analyses supporting the EPU continue to satisfy the applicable acceptance criteria. The increase in the trip setpoint is the result of a risk evaluation and results in an overall risk reduction.

   Therefore, the proposed change does not result in a significant reduction in a margin of safety.

D. Emergency Core Cooling Minimum Boron Concentration

In order to ensure subcriticality following a loss of coolant accident (LOCA), consistent with the EPU safety analyses, the required boron concentrations in the safety injection tanks (SITs) and refueling water tank (RWT) in the emergency core cooling system (ECCS) TSs are being changed from between 1720 and 2100 ppm to between 1900 and 2200 ppm.

The other boration-related TSs including shutdown margin, borated water sources, refueling operations, and special test exceptions TS are also being revised to address the increase in minimum and maximum boron concentration and prevent a conflict with the ECCS TS.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The changes to the SITs and RWT required boron concentrations are bounded by the assumptions in the EPU safety analyses. The proposed changes to the shutdown margin, boration water sources, refueling operations, and special test exceptions TSs to increase the minimum and maximum boron concentration have been evaluated and do not conflict with the ECCS TSs. Components and systems will continue to function as designed and performance requirements for these will continue to be satisfied.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The changes to the SITs and RWT required boron concentrations are bounded by the assumptions in the EPU safety analyses. No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed changes. SSCs previously required for mitigation of a transient continue to be capable of fulfilling their design functions. The proposed changes have no adverse affect on any safety-related system or component and do not change the performance or integrity of any safety-related system.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The proposed changes to the TS boron concentration requirements for the SITs and RWT and the other boration related TSs are consistent with the EPU safety analyses. The design basis limits for the accident and transient analyses are met at EPU conditions. Systems and components will continue to meet their design basis criteria.

   Therefore, the proposed changes do not result in a significant reduction in a margin of safety.

E. Emergency Core Cooling System Subsystems

A cross-reference is provided in the ECCS Subsystems – Operating LCO to alert the operator that the inoperability of a charging pump may impact the TS requirement for maintaining the required boron injection flow paths operable and the charging pumps are added to the surveillance requirements for safety injection signal testing.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The changes do not involve physical changes to the plant and do not adversely affect the operation of St. Lucie Unit 2. These are enhancements that provide further assurance that ECCS sources credited in the safety analyses are available. Components and systems will continue to function, performance requirements for these systems will continue to be satisfied, and no safety limits will be exceeded.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The changes do not involve physical changes to the plant. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed administrative changes. The changes have no adverse effect on any safety-related system and do not change the performance or integrity of any safety-related equipment.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   These changes do not involve physical changes to the plant, nor do they affect any safety analyses. They are enhancements that provide further assurance that ECCS sources credited in the safety analyses are available.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

F. Control Element Assembly Drop Time

The required time for a CEA dropped from it fully withdrawn position to 90% of its insertion limit is being increased.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change increases the required CEA drop time. This new CEA drop time requirement must be verified prior to Modes 1 or 2 of plant operations. The probability of an accident previously evaluated remains unchanged since the CEAs drop into the core as a result of a core anomaly or undesired condition, and the fact that the CEA drop time was increased does not in itself initiate an accident. Likewise, the consequences of an accident previously evaluated remain unchanged since for both LOCA and non-LOCA analyses, it has been verified that the proposed slower reactivity insertion rate at all rod positions, will not preclude meeting the trip reactivity limits used in the EPU analyses.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously analyzed.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The increase in CEA drop time as proposed in this TS change has been explicitly analyzed as part of the safety analyses under EPU conditions. These analyses have been shown to meet all of the applicable acceptance criteria. Thus, this change does not have any adverse effect on the existing margins of safety for the fuel, the fuel cladding, the reactor vessel, or the containment building. There is no impact on the mechanical design. The slightly slower drop would produce a smaller impact on the fuel assembly and lower stresses on the CEA. Since there is no adverse impact, current mechanical design analyses remain applicable.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

G. Reactor Coolant System Specific Activity

The $\bar{E}$ Average Disintegration Energy is being replaced with Dose Equivalent XE-133 and the TS rewritten to be consistent with TSTF-490. Accordingly, the waste gas storage tank TS is also revised to reflect the change to Dose Equivalent XE-133 and provide an associated activity limit for the tank(s).

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Reactor coolant specific activity is not an initiator for any accident previously evaluated. The proposed change will limit primary coolant activity to concentrations consistent with the accident analyses while the TS limit on the waste gas storage tank source term inventory is being reduced. With these limits, the EPU accident analyses continue to demonstrate that the resultant dose consequences remain within the applicable limits.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes in the limits on reactor coolant system (RCS) specific activity and the quantity of radioactivity in the waste gas storage tank do not alter any physical part of the plant nor affect any plant operating parameter.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The proposed changes revise the limits on noble gas radioactivity in the reactor coolant and radioactivity in the waste gas storage tank. The proposed changes are consistent with the assumptions in the safety analyses and will ensure the monitored values protect the initial assumptions in the safety analyses.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

H. Reactor Coolant System Pressure

The range of the lift settings of the pressurizer code safety valves is being increased.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The EPU safety analyses have been shown to support an increased range of tolerance on the as-found lift settings of the pressurizer code safety valves. With the increase in tolerance range, the pressurizer code safety valves, in conjunction with the secondary safety valves and the reactor protection system, will continue to maintain the RCS below design limits.

The operation of the pressurizer code safety valves is not affected by the proposed change. SSCs required to mitigate transients continue to be capable of performing their design functions. The applicable RCS pressure acceptance criteria continue to be satisfied. Accordingly, the proposed change does not increase the consequences of an accident.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The increased as-found pressurizer code safety valve setpoint tolerance range will not result in a new or different accident from any previously evaluated. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The applicable accident analyses demonstrated acceptable results assuming the opening setpoint of the pressurizer code safety valves was biased to the revised tolerance limits.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

I. Reactor Coolant System Pressure – Temperature Limits and Low Temperature Overpressure Protection (LTOP)

The applicability of the pressure-temperature (P-T) limit curves is being changed from 55 to 47 EFPY.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The only change to the P-T limit curves as a result of EPU is their period of applicability. The P-T limit curves have been determined in accordance with the methodologies set forth in the regulations to provide adequate margin of safety to ensure that the reactor vessel will withstand the effects of normal startup and shutdown cyclic loads due to system temperature and pressure changes as well as the loads associated with reactor trips. The regulations of 10 CFR 50 Appendix A, GDC-14 and -31 remain satisfied. The P-T limit curves in the TS are conservatively generated in accordance with the fracture toughness requirements of the ASME Code Section XI, Appendix G. The margins of safety against fracture provided by the P-T limits using the requirements of 10 CFR 50 Appendix G are equivalent to those recommended in ASME Section XI, Appendix G. The adjusted reference temperature values are based on the guidance of RG 1.99, Revision 2.

The proposed change in period of applicability will not result in physical changes to the SSCs or to event initiators or precursors.

The proposed change will not impact assumptions and conditions previously used in the radiological consequence evaluations, nor affect mitigation of these consequences due to accidents described in the UFSAR. Also, the proposed change will not impact a plant system, such that previously analyzed SSCs might be more likely to fail. The initiating conditions and assumptions for accidents described in the UFSAR remain as analyzed.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The requirements for P-T limits curves and LTOP have been in place since the beginning of plant operation. The P-T curves are based on reactor vessel irradiation damage predictions using RG 1.99, Revision 2 methodology. The only change to the P-T limit curves as a result of EPU is their period of applicability. No new failure modes are identified nor are any SSCs required to be operated outside of their design bases.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The P-T curves continue to maintain the safety margins of 10 CFR 50, Appendix G during the proposed revision to their period of applicability by defining the limits of operation which prevent non-ductile failure of the reactor pressure vessel. Analyses have demonstrated that the fracture toughness requirements are satisfied and that conservative operating restrictions are maintained for the purpose of LTOP. The P-T limit curves provide assurance that the RCS pressure boundary will behave in a ductile manner and that the probability of a rapidly propagating facture is minimized.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

J. Containment System – Internal Pressure

The peak calculated containment internal pressure for the design basis LOCA (P_a) is increased to 43.48 psig.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The peak calculated pressure for the design basis LOCA is less than the containment design pressure of 44 psig. The containment analyses demonstrate that containment systems will continue to provide sufficient pressure and temperature mitigation capability under EPU conditions to ensure that containment integrity is maintained.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the revised peak calculated containment internal pressure following the design basis LOCA. The containment analyses demonstrate that containment systems will continue to provide sufficient pressure and temperature mitigation capability under EPU conditions to ensure that containment integrity is maintained.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The containment analyses demonstrate that the containment and associated systems and components continue to satisfy the applicable design criteria.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

K. Main Steam Safety Valves

The main steam safety valve (MSSV) as-found upper setpoint tolerance limits are being increased to provide operating margin and to account for setpoint drift. The MSSV lift setpoints, the as-found lower setpoint tolerance limit and the as-left setpoint tolerance limits are unchanged for EPU.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The increase in the as-found upper setpoint does not impact the probability of any accident previously evaluated. The applicable accident analyses demonstrated acceptable results assuming the opening setpoint of the MSSVs was biased to the revised upper tolerance limits.

   The operation of the MSSVs is not affected by the proposed change. SSCs required to mitigate transients continue to be capable of performing their design functions. Acceptance criteria continue to be satisfied. Accordingly, the proposed change does not increase the consequences of an accident.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The increased as-found upper MSSV setpoint tolerance limit will not result in a new or different accident from any previously evaluated. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The applicable accident analyses demonstrated acceptable results assuming the opening setpoint of the MSSVs was biased to the revised upper tolerance limits.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

L. Electrical Power Sources – Fuel Oil Supply

The emergency diesel generator (EDG) minimum fuel oil volume requirement is being increased from 40,000 to 42,500 gallons. This change supports the use of ultra low sulfur fuel oil which has a lower heat rate (currently addressed by an administrative limit) and meets the fuel oil requirements for EDG loading under EPU conditions.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The EDGs fuel oil tanks are part of a system used to mitigate the consequences of an accident and do not increase the possibility of an accident previously evaluated.

   The increase in minimum fuel oil requirements enables operation of the EDGs to remain unchanged for EPU fuel consumption rates, thus the EDGs continue to be capable of performing their design functions. Acceptance criteria continue to be satisfied. Accordingly, the proposed change does not increase the consequences of an accident.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the increase in minimum fuel oil tank volume. The proposed change has no adverse effect on any safety-related system and does not change the performance or integrity of any safety-related equipment. No new safety-related equipment is being added or replaced as a result of the proposed change.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The EDG fuel consumption analyses demonstrate that the EDG design continues to satisfy its safety function. The design basis limits for the accident and transient analyses will continue to meet their design criteria.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

M. Electrical Power Sources – Voltage and Frequency

The EDG steady-state frequency tolerance is changed from +/- 2% to +/- 1% and the voltage tolerance is changed from +/- 10% to +/- 5%. The tightening of the tolerance for EDG voltage and frequency is more conservative than the current tolerances.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The EDGs are used to mitigate the consequences of an accident and are not accident initiators. Accordingly, the EDGs do not increase the possibility of an accident previously evaluated.

   The operation of the EDGs is unchanged for EPU. The impact on EDG-powered motor-operated valves (MOVs) and pumps has been evaluated for the worst case values of frequency and voltage and the SSCs continue to be capable of performing their design functions. Acceptance criteria continue to be satisfied. Accordingly, the proposed change does not increase the consequences of an accident.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of tightening the tolerance on EDG voltage and frequency. No new safety-related equipment is being added or replaced as a result of the proposed change.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The analyses continue to satisfy the acceptance criteria with respect to the EDG design. The design basis limits for the accident and transient analyses will continue to meet their design criteria.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

N. Design Features – Fuel Storage

The fuel storage TSs are being revised to address the replacement of the CLB analysis with an analysis (Appendix G to Attachment 5) that bounds both pre-EPU and EPU fresh and spent fuel with an initial enrichment of 1.5 to 4.6 weight percent U-235 and a fuel burnup for operation at 3020 MWt as well as an increase in the spent fuel pool storage rack capacity.

The existing figures are being replaced with new figures that reflect the fuel storage and fuel arrangement configurations in the spent fuel pool and cask pit storage areas that are allowable under the revised criticality analysis. A new table is also added that provides the inputs into the minimum burnup calculations for these allowed configurations and interfaces.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The fuel enrichment and burnup history are used to determine the consequences of an accident and are not accident initiators. Accordingly, the changes in fuel enrichment and burnup do not increase the possibility of an accident previously evaluated.

   A criticality analysis was conducted using the increased fuel enrichment and burnup. The analysis determined the allowable fuel storage and fuel arrangement patterns that will continue to satisfy the applicable acceptance criteria including a loss of soluble boron in the spent fuel pool event. These are reflected in the proposed revised TS.

   The new licensing basis criticality analyses, as well as the thermal and structural analyses, support the proposed increase in the TS limit for available spent fuel pool rack storage capacity.

   The new fuel design was evaluated and the EPU safety analyses continue to meet applicable acceptance criteria for dose consequences.

   Therefore, the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The fuel enrichment and burnup history are used to determine the consequences of an accident, and will not affect the design functions of any systems or components. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change.

   The proposed change has no adverse effect on any safety-related system and does not change the performance or integrity of any safety-related equipment. No new safety-related equipment is being added or replaced as a result of the proposed change.

Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The analyses demonstrate that the applicable acceptance criteria will continue to be satisfied with respect to reactivity limits and fuel design.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

O. Core Operating Limits Report (COLR) References

These changes are administrative in nature. The changes update the COLR reference list to the latest applicable documents.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   These administrative changes update the list of references used to support the reload safety analysis that is provided in the TSs and in the COLR. The changes do not involve any physical changes to the plant and do not affect the operation of St. Lucie Unit 2. Components and systems will continue to function, performance requirements for these systems will continue to be satisfied, and no safety limits will be exceeded.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The changes to the COLR reference list are administrative in nature. The changes do not involve any physical changes to the plant. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed administrative changes. The changes have no adverse effect on any safety-related system and do not change the performance or integrity of any safety-related equipment.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The analyses continue to satisfy the acceptance criteria with respect to the fuel design. The administrative changes do not affect any safety analyses.

   Therefore, the proposed TS change does not involve a significant reduction in a margin of safety.

P. Dose Analyses

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The determination of dose is not an accident initiator. Accordingly, the dose calculations do not increase the probability of an accident previously evaluated.

   The alternative source term methodology was used to determine the consequences of design basis accidents (DBAs). Doses were calculated for the exclusion area boundary (EAB), low population zone (LPZ), and control room for each accident. Onsite and offsite dose consequences are the result of postulated accidents and are not accident initiators. Accordingly, the dose calculations and the associated results do not increase the probability of an accident previously evaluated.

   Using conservative, bounding assumptions for key inputs such as control room unfiltered inleakage and fuel centerline melt, the calculated doses remain less than the limits established in 10 CFR 50.67, Accident Source Term, and the acceptance criteria contained in RG 1.183, BTP 11-5, and GDC-19.

   Therefore, this dose does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Dose calculations are used to determine the consequences of an accident, and are not accident initiators. The change does not involve the addition or modification of equipment, not does it alter the design or operation of plant systems.

   Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The calculated doses remain less than the limits established in 10 CFR 50.67, Accident Source Term, and the acceptance criteria contained in RG 1.183, BTP 11-5, and GDC-19.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed EPU amendment to the St. Lucie Unit 2 RFOL, TS and CLB presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

5.3 Environmental Evaluation

The environmental considerations evaluation is contained in Attachment 2, Supplemental Environmental Report. It concludes that EPU will not result in a significant change in non-radiological impacts on land use, water use, waste discharges, terrestrial and aquatic biota, transmission facilities, or social of economic factors, and will not have non-radiological environmental impacts other than those evaluated in the Environmental Report. The Environmental Report concludes that EPU will not introduce any new radiological release pathways, will not result in a significant increase in occupational or public radiation exposures, and will not result in significant additional fuel cycle environmental impacts.

FPL has determined that operation with the proposed EPU license amendment would not result in any significant change in the types or significant increase in the amounts of effluent that may be released offsite nor does it involve a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed license amendment is eligible for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 50.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed license amendment.

6.0 REFERENCES

1. NRC Review Standard RS-001, Review Standard for Extended Power Uprate, U. S. Nuclear Regulatory Commission, December 2003.
2. NRC Regulatory Issue Summary 2002-03, Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications, January 31, 2002.
3. Environmental Protection Agency Federal Guidance Report No. 12 (FGR 12), External Exposure to Radionuclides in Air, Water, and Soil, 1993.
4. Technical Specification Task Force (TSTF)-490 Revision 0, Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec, September 13, 2005.
5. NUREG-1432, Revision 3.0, Standard Technical Specifications Combustion Engineering Plants, June 2004.
6. NRC RIS 2006-17, NRC Staff Position on the Requirements of 10 CFR 50.36, “Technical Specifications,” Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels, August 24, 2006.
7. P.L.Hiliard, Chief Reactor Operations Branch, NRC letter to Nuclear Energy Institute, Technical Specification For Addressing Issues Related To Setpoint Allowables, September 7, 2005.
8. NRC Regulatory Guide (RG) 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, March 2001.
9. NRC Regulatory Guide.(RG) 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, May 1988.
10. NRC Regulatory Guide 1.183, Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants, July 2000.
11. NUREG-0800 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.
12. NRC letter to FPL, St. Lucie Plant, Unit No. 2 - Issuance of Amendment Regarding Change in Reload Methodology and Increase in Steam Generator Tube Plugging Limit, January 31, 2005.
13. NRC letter to FPL, St. Lucie Plant, Unit No. 2 – Issuance of Amendment for Reduction in Reactor Coolant System Flow and Increase in Steam Generator Plugging Limit, May 2006.