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 NUCLEAR PLANT, UNIT 1

This coversheet plus 47 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 (FOL) No. DPR-67 for St. Lucie Nuclear Plant Unit No. 1 (St. Lucie Unit 1).

The proposed license amendment request (LAR) will revise the FOL to permit St. Lucie Unit 1 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). Once approved, the amendment implementation is planned following the Fall 2011 refueling outage. Power ascension to EPU power level is planned to commence following NRC approval of the EPU LAR and completion of the Fall 2011 refueling outage. There are no other St. Lucie Unit 1 LARs that are required for implementation of the EPU.

FPL has reviewed the FOL, 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 FOL 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 9, 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 1 TS. Changes that are not required to support EPU are identified as such.

2.0 BACKGROUND

St. Lucie Unit 1 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 1 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 1. The results of this evaluation are described in LAR Attachment 5, Licensing Report (LR). The EPU LR supports the requested EPU changes to the FOL, 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 FOL, TS, and CLB. There are no other St. Lucie Unit 1 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 FOL, 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. FOL DPR-67, CONDITION 3.A - MAXIMUM POWER LEVEL

   - The maximum reactor core power level is revised from “2700 megawatts (thermal)” to “3020 megawatts (thermal).”

   Basis for the Change: 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.

2. INDEX

   DEFINITIONS

   - “1.11 $\overline{E}$ 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.”

   LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

   - “3/4.1.2.8 Borated Water Sources – Operating” page number is changed from “3/4 1-18” to “3/4 1-17”,

- “3/4.9.7 Crane Travel – Spent Fuel Storage Pool Building” is changed to “3/4.9.7 DELETED”,
- “3/4.9.13 Spent Fuel Storage Cask Crane” is changed to “3/4.9.13 DELETED”, and
- “3/4.9.14 Decay Time - Storage Pool” is changed to “3/4.9.14 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 new 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 RCS OVERPRESSURE PROTECTION RANGE

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

Basis for the Change: The deletion of this definition is administrative, since the actual values for the low temperature overpressure protection (LTOP) settings are provided in LCO 3.4.13 Reactor Coolant System – Power Operated Relief Valves. This change is consistent with NUREG-1432 Standard Technical Specifications Combustion Engineering Plants (Reference 5) which does not contain a definition for Low Temperature RCS Overpressure Protection Range.

5. TS 1.25, DEFINITIONS - RATED THERMAL POWER

- The RATED THERMAL POWER (RTP) 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 – FOUR REACTOR COOLANT PUMPS OPERATING

- Replacement for FIGURE 2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT – FOUR REACTOR COOLANT PUMPS OPERATING, and
- The "VESSEL FLOW LESS MEASUREMENT UNCERTAINTIES – 365,000 GPM" is changed to "REACTOR COOLANT SYSTEM TOTAL FLOW RATE – SPECIFIEC IN COLR Table 3.2-1" in the replacement figure.

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

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. Overheating of the fuel cladding is prevented 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 maximum cold leg 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 incorporate additional data points from the revised EPU analysis.

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, SSCs, accidents, and fuel analyses and evaluations. One of the major inputs and assumptions used in the calculation of the NSSS design parameters established an increased minimum RCS total flow requirement of 375,000 gpm to ensure that the reactor core thermal margin safety limit is not exceeded.

Included in the revised TS Figure 2.1-1, the RCS total flow requirement note is changed to indicate that it is now provided in COLR Table 3.2-1. The minimum reactor coolant flow is determined for each refueling cycle core in the reload safety analysis. The minimum required reactor coolant flow value with four pumps operating is determined and provided in the COLR. This change ensures that the reactor coolant flow requirement is consistent with the value determined in the reload safety analysis.

Relocating the specific parameter limits to the COLR is consistent with NUREG-1432 and the St. Lucie Unit 2 Technical Specifications. The relocation to the COLR is not required for EPU and is included as a TS enhancement.

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

- Table 2.2-1, Functional Unit in the Table for "7. Steam Generator Water Level – Low" is changed to "7. Steam Generator Water Level – Low**,***" to include two new notes as described below,
- Table 2.2-1, Trip Setpoint for low steam generator level in Table 2.2-1 is changed from "≥ 20.5% Water Level – each steam generator" to "≥ 35.0% Water Level – each steam generator",
- Table 2.2-1, Allowable Value for low steam generator level in Table 2.2-1 is changed from "≥ 19.5% Water Level – each steam generator" to "≥ 34.78% Water Level – each steam generator",

- Table 2.2-1, the current footnote is changed from "* Design reactor coolant flow with 4 pumps operating is 365,000 gpm." to "* For minimum reactor coolant flow with 4 pumps operating, refer to COLR Table 3.2-1",
- Table 2.2-1, a new note is added below Table 2.2-1 that reads "** 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", and
- Table 2.2-1, a new note is added below Table 2.2-1 that reads "*** 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".

<u>Licensing Report:</u> 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 2.8.5.0, Accidents and Transient Analysis; Section 2.13, Risk Evaluation; and Appendix E, Supplement to Licensing Report; Section 2.4.1, Reactor Protection Safety Features Actuation, and Control Systems; and Section 2.4.2, Plant Operability

<u>Basis for the Change:</u> 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 for reactor trip on low steam generator level at 20.5% of steam generator level with all RCPs operating, and show that the results are within acceptable safety limits. Hence, this TS change is not related to the EPU increase in power level. As described below, this TS change 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 increase of the reactor trip on low SG level to ≥ 35.0% and revision of plant procedures that provide for the trip of all RCPs following the identification of a total loss of feedwater events, operator action times for restoration of feedwater and transfer to once through cooling would be increased following EPU from those action times for the current licensed power level. Hence, the implementation of the increase in the reactor trip set point to ≥ 35.0% 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 established an increased minimum RCS total flow requirement of 375,000 gpm 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 footnote (*) for TS Table 2.2-1 is being revised to change “design reactor coolant flow” to “minimum reactor coolant flow” and remove the specific value for reactor coolant flow and to refer to the COLR. The change from “design reactor coolant flow” to “minimum reactor coolant flow” provides a more accurate description of the actual parameter determination. This change ensures that the reactor coolant flow requirement is consistent with the value determined in the reload safety analysis.

Relocating the specific parameter limits to the COLR is consistent with NUREG-1432 and the St. Lucie Unit 2 Technical Specifications. The relocation to the COLR is not required for EPU and is included as a TS enhancement.

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, and required actions were added to the TS should the setpoint be outside the Allowable Value limits. Two new footnotes are added to the bottom of TS Table 2.2-1 for the RPS trip on low steam generator level. These footnotes are consistent with the two recommended notes provided in 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} > 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 – $T_{avg} \leq$ 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 2.8.5.0, Accident and Transient Analysis

   Basis for the Change: A sufficient shutdown margin ensures that the reactor can be made subcritical from all operating conditions, the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 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 revised boron concentration is consistent with the increased boron concentration required in the refueling water tank (RWT) and the safety injection tanks (SITs) and can also be provided by the boric acid makeup (BAM) tanks.

9. TS 3/4.1.2.1, REACTIVITY CONTROL SYSTEMS – FLOW PATHS – SHUTDOWN

   • LCO 3.1.2.1b contains a footnote designated by an asterisk (*). The second sentence of this footnote which starts, "In the latter case:…" is being revised as follows:

     - "In the latter case: 1) all … their power removed:" is changed to "In the latter case, all charging pumps shall be disabled.", and

     - The list of valves in footnote (*) is deleted.

   • Figure 3.1-1b is deleted.

   TS 3/4.1.2.3, REACTIVITY CONTROL SYSTEMS – CHARGING PUMPS - SHUTDOWN

   • LCO 3.1.2.3 contains a footnote designated by an asterisk (*). The second sentence of this footnote which starts, "In the latter case:…" is being revised as follows:

     - "In the latter case: 1) all … their power removed:" is changed to "In the latter case, all charging pumps shall be disabled", and

     - The list of valves in footnote (*) is deleted.

   <u>Licensing Report:</u> Section 2.1.2, Pressure – Temperature Limits and Upper Shelf Energy

   <u>Basis for the Change:</u> The noted "(*)" provisions are provided in the current TS to assure that RCS overpressurization does not occur while maintaining these boration control flow paths. The LTOP system is designed to prevent RCS overpressurization above the 10 CFR 50 Appendix G pressure-temperature (P/T) limits curves, based on revised analysis conducted with EPU power levels as appropriate. The TS revisions to update the P/T limits curves and revise the LTOP enabling temperatures are discussed in Changes Nos.14 and 15 below.

   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 are compared to the RCS P/T limits. The results of the analyses demonstrate that the allowable heatup rates and LTOP enable temperatures can be maintained for heatup. The analysis to generate the new P/T limits curves has determined that there is sufficient margin that when using the HPSI pump as the required boron injection flow path, it is not necessary to limit the HPSI pump flow by closing and deactivating two header isolation valves. Hence, TS Figure 3.1-1b is no longer needed, and is deleted.

10. 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.

11. 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 3650 gallons of 2.5 to 3.5 weight percent boric acid (4371 to 6119 ppm boron).” to “a minimum borated water volume of 3650 gallons of 3.0 to 3.5 weight percent boric acid (5245 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 FIGURE 3.1-1, ST. LUCIE 1 MIN BAMT VOLUME VS STORED BAMT CONCENTRATION – This figure is replaced with new TS Figure 3.1-1, entitled “FIGURE 3.1-1, ST. LUCIE 1 MIN BAMT VOLUME VS STORED BAMT CONCENTRATION (Modes 1, 2, 3 and 4)”, and
- TS Figure 3.1-1 is relocated to TS 3/4.1.2.8.

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

- LCO 3.1.2.8.d.2. The RWT minimum boron concentration is changed from “1720 ppm” to “1900 ppm”,
- Pages 3/4.1-17, 3/4.1-18, and 3/4.1-19 are renumbered to reflect the relocation of Figure 3.1-1 to TS 3/4.1.2.8.

Licensing Report: Section 2.8.5.0, Accident and Transient Analysis

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 RWT and BAM tanks minimum boron concentrations are increased to ensure that the 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.

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. TS Figure 3.1-1 is being relocated from this specification to TS 3/4.1.2.8 Borated Water Sources – Operating. LCO 3.1.2.8 requires BAM tanks to meet the requirements of TS Figure 3.1-1 in Modes 1, 2, 3 and 4. LCO 3.1.2.7 provides the specific values for the BAM tanks in Modes 5 and 6. Accordingly, it is more appropriate to include TS Figure 3.1-1 with TS 3.1.2.8. The relocation of TS Figure 3.1-1 is a conforming change.

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 3.2-1”,

- 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 3.2-1 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 footnote originally at the bottom of Table 3.2-1 is added to the bottom of the page 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 at the bottom on the page is revised from “*” to “**”, and
- Table 3.2-1 “DNB MARGIN LIMITS” is deleted from TS and relocated to the COLR.

<u>Licensing Report:</u> Section 1.1, Nuclear Steam Supply System (NSSS) Parameters

<u>Basis for the Change:</u> The limits on the departure from nucleate boiling (DNB) related parameters ensure that each of the parameters are 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.

The DNB margin limit for cold leg temperature is changed from ≤ 549°F to ≤ 551°F and reactor coolant flow rate is changed from ≥ 365,000 gpm to ≥ 375,000 gpm. The revised limits on the DNB-related parameters assure that the parameters are maintained within the normal steady-state envelop of operation assumed in the accident and transient analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater that or equal to the DNBR limit throughout the transient analyses.

TS Table 3.2-1 is being relocated to the COLR. The DNB parameters are determined for each refueling cycle core in the reload safety analysis. This change ensures that the DNB parameters are consistent with the values determined in the reload safety analysis.

Relocating the asterisk “*” from the limiting value “≥ 2225 psia” to the parameter “Pressurizer Pressure” is an administrative change to ensure that the requirements of the footnote are maintained when moving the value to the COLR.

Relocating the specific parameter limits to the COLR is consistent with NUREG-1432 and the St. Lucie Unit 2 Technical Specifications. The relocation to the COLR is not required for EPU and is included as a TS enhancement.

The footnote is revised to correct an administrative error from a previous TS revision. The words “per minute” were inadvertently omitted from the previous revision and are required to provide an accurate description of a ramp increase. This is an administrative change.

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

- LCO 3.4.8.b, is changed from “$\leq 100/\overline{E}$ μCi/gram.” to “≤ 518.9 μCi/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 ≤ 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, Item 1 is changed as follows:
  - TYPE OF MEASUREMENT AND ANALYSIS is changed from “Gross Activity Determination” to “DOSE EQUIVALENT XE-133 Determination.”
  - MINIMUM FREQUENCY is changed from “3 times per 7 days with a maximum time of 72 hours between samples” 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.”
  - MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED is changed from “1#, 2#, 3#, 4# and 5#” to “1#, 2#, 3# and 4#”,

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 those isotopes, based on low concentration, short half-life, or small dose conversion factors, and 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.

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.

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 "3 times per 7 days with a maximum time of 72 hours between samples" to "1 per 7 days." 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.

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.

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

- TS 3.4.9.1 LCO is changed from "…Figures 3.4-2a, 3.4-2b and 3.4-3 …" to "… Figure 3.4-2a and Figure 3.4-2b …",
- APPLICABILITY: is changed from "At all times. *#" to "At all times. *",

- TS 3.4.9.1 ACTION is changed from "…Figures 3.4-2b and 3.4-3." to Figure 3.4-2b.",
- The current asterisk (*) footnote related to APPLICABITY is deleted,
- The current (#) footnote related to APPLICABILTY is retained but changed to (*),
- Current SURVEILLANCE REQUIREMENT 4.4.9.1.c "… shall be used to update Figures 3.4-2a, 3.4-2b and 3.4-3." is changed to "… shall be used to update Figures 3.4-2a and 3.4-2b.",
- TS Figure 3.4-2a, ST. LUCIE UNIT 1 P/T LIMITS, 35 EFPY - HEATUP AND CORE CRITICAL is replaced with TS Figure 3.4-2a, ST. LUCIE UNIT 1 P/T LIMITS, 54 EFPY - HEATUP AND CORE CRITICAL,
- TS Figure 3.4-2b, ST. LUCIE UNIT 1 P/T LIMITS, 35 EFPY - COOLDOWN AND INSERVICE TEST is replaced with TS Figure 3.4-2b, ST. LUCIE UNIT 1 P/T LIMITS, 54 EFPY - COOLDOWN AND INSERVICE TEST, and
- TS Figure 3.4-3, ST. LUCIE UNIT 1 P/T LIMITS, 35 EFPY – MAXIMUM ALLOWABLE COOLDOWN RATES is deleted.

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

Basis for the Change: The revised P/T limits curves and LTOP analysis to 54 effective full power years (EFPY) are based on current ASME methods and regulatory requirements. The new curves incorporate instrument uncertainty margin.

The basis for the proposed change is provided in WCAP-17197, which is included as Appendix G to Attachment 5 of this LAR. The WCAP provides a compilation of the calculation notes for the P/T limits curves, LTOP analysis, and supporting calculations and evaluations. The fluence projections were prepared using the guidance of NRC RG 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (Reference 8). The reactor pressure vessel beltline 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).

15. TS 3/4.4.13, REACTOR COOLANT SYSTEM – POWER OPERATED RELIEF VALVES

- LCO 3.4.13.a is changed to delete the colon after "selected" and add "during heatup, cooldown and isothermal conditions when the temperature of any RCS cold leg is less than or equal to 200°F.",
- LCO 3.4.13.a.1. is deleted,
- LCO 3.4.13.a.2. is deleted,
- LCO 3.4.13.b is changed to delete the colon after "selected" and add "during heatup, cooldown and isothermal conditions when the temperature of any RCS cold leg is greater than 200°F and less than or equal to 300°F.",
- LCO 3.4.13.b.1. is deleted,
- LCO 3.4.13.b.2. is deleted, and

- LCO APPLICABILITY is changed from "…RCS cold leg is less than or equal to 304°F,…" to "…RCS cold leg is less than or equal to 300°F,…".

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

Basis for the Change: The LTOP system is designed to prevent violation of the RCS P/T limits in the event of an overpressure event during low temperature operation. The revised P/T limits curves are described in Change No. 14 above. The design basis mass addition transients during low temperature operation are based on the operability requirements for the PORVs, the charging pumps, and the 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 are compared to the RCS P/T limits.

The results of the analyses demonstrate that the allowable heatup rates and LTOP enable temperatures can be maintained for heatup. However, the PORV setpoint transitions are being revised as indicated above to provide additional margin during RCS heatup and cooldown, and increased operational simplicity.

16. TS 3/4.4.14, REACTOR COOLANT SYSTEM – REACTOR COOLANT PUMP STARTING

- LCO 3.4.14 Note "#" at the bottom of page is changed from "# Reactor Coolant System Cold Leg Temperature is less than 304°F." to "# Reactor Coolant System Cold Leg Temperature is less than 300°F."

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

Basis for the Change: The noted, "(#)" provisions are provided in the current TS to assure that RCS overpressurization does not occur while in Mode 4 when RCS temperature is below the analyzed value of 300°F without an RCP operating. The LTOP system is designed to prevent RCS overpressurization above the 10 CFR 50 Appendix G P/T limits curves at RCS temperatures at or below 300°F during heatup. The TS revisions to update the P/T limits curves and revise the LTOP enabling temperatures are discussed in Changes Nos. 14 and 15 above.

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

- LCO 3.5.1.c. The SIT minimum boron concentration is changed from "1720 PPM" to "1900 ppm", and

- LCO 3.5.1.d. The SIT nitrogen cover-pressure is changed from "between 200 and 250 psig" to "between 230 and 280 psig."

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

Basis for the Change: 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 SIT nitrogen cover-pressure range is increased to provide borated water into the reactor at a higher RCS pressure following a loss-of-coolant accident (LOCA). The pre-EPU minimum SIT tank minimum pressure of 200 psig and the higher tank pressure of 280 psig were used in the realistic large break LOCA (RLBLOCA) analysis, as required, and the results are within acceptable limits. However, for the worst case SBLOCA, the minimum pressure for the SITs of 230 psig was required under EPU conditions. Hence, the revised range in the TS includes this higher minimum pressure requirement of 230 psig.

18. TS 3/4.5.2, EMERGENCY CORE COOLING SYSTEMS (ECCS) – OPERATING

- LCO 3.5.2.b. The word “and” is deleted from the end of this LCO requirement,
- LCO 3.5.2.c. The end of this LCO is revised to delete the period and add “, and”,
- LCO 3.5.2.d. This is a new requirement stating “One OPERABLE charging pump*.”,
- APPLICABILITY is changed from “MODES 1, 2 and 3*” to “MODES 1, 2 and 3**”.
- Footnote “* With pressurizer pressure ≥ 1750 psia.” is changed to “** With pressurizer pressure ≥ 1750 psia.”
- 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.”
- Surveillance Requirement 4.5.2.e.1. is revised from “… valve in the flow path actuates …” to “… valve in the flow paths actuates …”,
- Surveillance Requirement 4.5.2.e.2.a. is revised from “High-Pressure Safety Injection Pump.” to “High-Pressure Safety Injection Pumps.”,
- Surveillance Requirement 4.5.2.e.2.b. is revised from “Low-Pressure Safety Injection Pump.” to “Low-Pressure Safety Injection Pumps.”,
- A new Surveillance Requirement 4.5.2.e.2.c. “Charging Pumps.” is added,
- Surveillance Requirement 4.5.2.f. an underline “_” is deleted.

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

Basis for the Change: 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 charging pumps are credited as an ECCS subsystem in the analysis of the SBLOCA. Hence, the charging pumps are added to the LCO to ensure availability of required equipment. Minor corrections are also made to the safety injection signal testing surveillances to assure that it is clear that flow paths rather than a 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 minimum boron concentration is changed from “1720 ppm” to “1900 ppm.”

Licensing Report: Section 2.8.5.6.3, Emergency Core Cooling System and Loss-of-Coolant Accidents and Attachment 5 Appendix C, Realistic Large Break LOCA Summary Report

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 the reactor will remain 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 RWT minimum boron concentration is increased to ensure that the assumptions for RWT injection used in the safety analyses are met.

20. TS 3/4.6.1.4, CONTAINMENT SYSTEMS – INTERNAL PRESSURE

- LCO 3.6.1.4 The primary containment internal pressure upper limit is changed from “2.4 PSIG” to “+ 0.5 psig.”

Licensing Report: Section 2.6.1, Containment Functional Design and Section 2.7.7, Other Ventilation Systems (Containment)

Basis for the Change: Containment vessel integrity ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. The limitations on containment internal pressure ensure that the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere and the containment peak pressure does not exceed the design pressure during design basis accident conditions.

The revised containment pressure LCO of + 0.5 psig for initial positive containment pressure will limit the total pressure following a design basis LOCA and a steam line break inside containment to less than 44.0 psig, which is the design pressure and is consistent with the accident analyses. The revised containment pressure limit of + 0.5 psig is more restrictive than the current limit of 2.4 psig.

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

- Table 4.7-1 STEAM LINE SAFETY VALVES PER LOOP
  - Change column header from “LIFT SETTING (+1% to -3%) to “LIFT SETTING*”,
  - Change the upper limit for valves 8201, 8202, 8203, 8204, 8205, 8206, 8207, and 8208 from “≤ 995.3 psig” to “≤ 1015.3 psig,”
  - Change the upper limit for valves 8209, 8210, 8211, 8212, 8213, 8214, 8215, and 8216 from “≤ 1035.7 psig” to “≤ 1046.1 psig,” and
  - A new Footnote is added to read, “* ± 3% for valves a through d and +2%/-3% for valves e through h”.

Licensing Report: Section 2.8.5.2.1, 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 4.7-1, are based on lift tolerances of +1/-3% for all valves. The positive setpoint tolerance for both banks of safety valves 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.

22. TS 3/4.7.1.3 PLANT SYSTEMS – CONDENSATE STORAGE TANK

- LCO 3.7.1.3. The condensate storage tank minimum contained volume is changed from “116,000 gallons” to “153,400 gallons.”

Licensing Report: Section 2.5.4.5, Auxiliary Feedwater System

Basis for the Change: The condensate storage tank (CST) minimum water volume is revised to ensure that sufficient water is available to accommodate decay heat removal for 7.66 hours, which includes maintaining Hot Standby for one hour and then performing a cooldown of the RCS to less than 325°F in the event of a total loss of offsite power. The current TS for minimum contained volume does not include an allowance for water that is not usable. Administrative controls currently in place, maintain the CST usable volume above the TS limit of 116,000 gallons. For EPU, the total required CST volume to maintain the required usable volume of 130,500 gallons is 153,400 gallons, which includes an allowance for water not usable because of the tank outlet nozzle location, submergence level above the outlet nozzle to prevent vortexing, and instrument uncertainty. The effect of auxiliary feedwater pump heat on the temperature of the fluid and the potential adverse effect on the required CST inventory have been evaluated.

In addition, St. Lucie Unit 1 credits the Unit 2 CST to provide an additional volume of water to support the Unit 1 analyses. The current Unit 2 TS requirement ensures a sufficient volume of water to support both the Unit 2 requirements and the Unit 1 EPU requirements. The Unit 2 TS Bases will be revised to address the change in water volume for the Unit 1 EPU analyses.

23. TS 3/4.8.1, ELECTRICAL POWER SYSTEM – AC SOURCES

TS 3/4.8.1.1 A.C. SOURCES – OPERATING

- LCO 3.8.1.1.b.2. The minimum fuel storage is changed from "… 16,450 gallons …" to "… 19,000 gallons …",
- Surveillance Requirement 4.8.1.1.2.e.3.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.4. This requirement is rewritten as indicated below:

  4 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.5.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.6. This requirement is rewritten as indicated below:

  6 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 3960 kW#, and

d) During the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3300 to 3500 kW#.

TS 3/4.8.1.2 A.C. SOURCES – SHUTDOWN

- LCO 3.8.1.2.b.2. The minimum fuel storage is changed from "… 16,450 gallons …" to "…19,000 gallons …"

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: A revised calculation of the minimum diesel oil fuel storage requirement has been completed, and the minimum fuel oil storage tank volume is being changed from 16,450 gallons to 19,000 gallons. The increased fuel oil storage is the result of using ultra low sulfur fuel oil with an accompanying lower heat rate. St. Lucie Unit 1 currently carries an administrative limit on the diesel fuel oil storage requirement. A revised loading calculation has been determined that the EPU loads result in a small increase in the fuel oil requirement. The increase to 19,000 gpm provides sufficient fuel to support the EPU loads and the ultra low sulfur fuel oil requirement.

The revised steady state frequency, 60 ± 0.6 Hz (± 1%), and revised steady state voltage, 4160 ± 210V (± 5%), reflect worst case values used in determining motor-operated valve (MOV) and pump loads connected to an emergency diesel generators (EDG). The accident analyses considered the over and under frequency and voltage uncertainty tolerance of the EDGs.

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

For MOVs with AC motors powered from the EDGs, the voltage tolerance of ± 5% will not affect the motor speed, 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 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 very low significance and treated as a non-cited violation.

The proposed change to these TS that tightens the steady state frequency and voltage tolerances is conservative, and is an enhancement to the St. Lucie Unit 1 TS. The changes associated with voltage and frequency tolerances are not required for EPU.

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

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

Licensing Report: Section 2.8.5.0, Accident and Transient Analysis

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 minimum boron concentration is increased from 1720 ppm to 1900 ppm to ensure that the core will remain subcritical during refueling operations involving core alterations or positive reactivity changes.

25. 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: Section 2.8.6.2, Spent Fuel Storage

Basis for the Change: 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.

The spent fuel storage pool minimum boron concentration is increased from 1720 ppm to 1900 ppm to ensures that the core will remain subcritical whenever irradiated fuel assemblies are in the spent fuel storage pool. The minimum concentration is consistent with the minimum boron concentration for the RWT, and assures an additional subcritical margin to the value of K_{eff} calculated in the spent fuel storage pool criticality analysis.

26. TS 3/4.9.14, REFUELING OPERATIONS – DECAY TIME – STORAGE POOL

- TS 3/4.9.14, This specification is deleted.

This is not an EPU required change.

Basis for the Change: The LCO for this TS prohibits movement of the spent fuel cask into the fuel cask compartment of the spent fuel storage pool until the specified decay times have been met. The basis for this restriction is to prevent a cask drop from threatening the structure of the storage pool. The installation of a single-failure proof crane eliminated this event from the current licensing basis. This is a conforming change that addresses previously approved TS Amendment 190 (Reference 10). As stated below, the TS does not meet any of the criteria specified in 10 CFR 50.36, therefore, it is not required and can be deleted. The deletion of this TS is consistent with NUREG-1432, and is considered to be an enhancement to the St. Lucie Unit 1 TS.

Comparison to 10 CFR 50.36 Criteria:

Criterion 1: *Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.*

Comparison: Load restriction on the spent fuel cask cranes are not related to any installed instrumentation that is used to detect reactor coolant pressure boundary degradation to indicate such degradation in the control room. Therefore, TS 3/4.9.14 does not meet Criterion 1.

Criterion 2: *A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.*

Comparison: As discussed in NRC Safety Evaluation to St. Lucie Unit 1 TS Amendment 190, FPL has installed single-failure proof cranes and revised the UFSAR to remove the cask drop accident from the licensing basis. Accordingly, Criterion 2 no longer applies.

Criterion 3: *A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that wither assumes the failure or presents a challenge to the integrity of a fission product barrier.*

Comparison: The systems, equipment, and criteria in this TS are not part of the primary success path that functions or actuates to mitigate a design basis accident or transient that either assumes the failure or presents a challenge to the integrity of a fission product barrier. Therefore, TS 3/4.9.14 does not meet Criterion 3.

Criterion 4: *A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.*

Comparison: The systems, equipment, and criteria contained in this TS are not components listed as risk significant under either the probabilistic risk assessment program or the maintenance rule program. Therefore, TS 3/4.9.14 does not meet Criterion 4.

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

- ACTION a. is changed from "…continue boration at ≥ 40 gpm of ≥ 1720 ppm boric acid solution" to "…continue boration at ≥ 40 gpm of ≥ 1900 ppm boric acid solution", and
- ACTION b. is changed from "…continue boration at ≥ 40 gpm of ≥ 1720 ppm boric acid solution" to "…continue boration at ≥ 40 gpm of ≥ 1900 ppm boric acid solution."

Licensing Report: Section 2.8.5.0, Accident and Transient Analysis

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 CEAs 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 boron concentration of 1900 ppm 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.

28. 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.

The waste gas decay tank inventory source term required to generate an exclusion area boundary dose of 0.1 rem 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 11-5, Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure, of Standard Review Plan Chapter 11, Radioactive Waste Management, of NUREG-0800 (Reference 12). 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 to of 0.1 rem TEDE is the basis for a proposed Technical Specification 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.134, TS 3/4.4.8, REACTOR COOLANT SYSTEM – SPECIFIC ACTIVITY, above.

29. TS 5.3, DESIGN FEATURES – REACTOR CORE – FUEL ASSEMBLIES

- TS 5.3.1, Fuel Assemblies, the entire paragraph is replaced with the following:

> The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods and/or poison rods, with fuel rods having an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies

that have not completed representative testing may be placed in non-limiting core regions.

Licensing Report: Section 2.8.1, Fuel System Design

Basis for the Change: The description of fuel assemblies is changed to provide a more accurate description and is consistent with the level of detail in NUREG-1432 and St. Lucie Unit 2 TS. The change allows for flexibility in fuel cladding, fuel, and core designs using NRC-approved materials, codes, and methods, without requiring pre-NRC approval. Core designs are evaluated for each operating cycle in the reload safety analysis.

30. TS 5.4.2, DESIGN FEATURES – REACTOR COOLANT SYSTEM – VOLUME

- TS 5.4.2 This specification is deleted.

This is not an EPU required change.

Basis for the Change: As stated below, this TS does not meet any of the criteria specified in 10 CFR 50.36, therefore, it is not required and can be deleted. In addition, the deletion of this TS is consistent with NUREG-1432 and St. Lucie Unit 2 TS, and is considered to be an enhancement to the St. Lucie Unit 1 TS.

Comparison to 10 CFR 50.36 Criteria:

Criterion 1: *Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.*

Comparison: RCS total water and steam volume is not related to any installed instrumentation that is used to detect reactor coolant pressure boundary degradation to indicate such degradation in the control room. Therefore, TS 5.4.2 does not meet Criterion 1.

Criterion 2: *A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.*

Comparison: RCS total water and steam volume is not related to any process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, TS 5.4.2 does not meet Criterion 2.

Criterion 3: *A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that wither assumes the failure or presents a challenge to the integrity of a fission product barrier.*

Comparison: RCS total water and steam volume is not related to any systems, equipment, and criteria that are not part of the primary success path that functions or actuates to mitigate a design basis accident or transient that either assumes the failure or presents a challenge to the integrity of a fission product barrier. Therefore, TS 5.4.2 does not meet Criterion 3.

Criterion 4: *A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.*

Comparison: RCS total water and steam volume is not related to any systems, equipment, and criteria listed as risk significant under either the probabilistic risk assessment program or the maintenance rule program. Therefore, TS 5.4.2 does not meet Criterion 4.

31. TS 5.6, DESIGN FEATURES – FUEL STORAGE - CRITICALITY

- TS 5.6.1.a.4. is changed from "… Criteria 1 and 3." to "… Criteria in 5.6.1.a.1 and 5.6.1.a.3",
- TS 5.6.1.a.6. is changed from "… Criteria 1 and 3." to "… Criteria in 5.6.1.a.1 and 5.6.1.a.3",
- TS 5.6.1.c. is changed from "… Criteria 2 and 3." to "… Criteria in 5.6.1.c.2 and 5.6.1.c.3",
- TS 5.6.1.c.1 is changed from "…less than or equal to 4.5 weight percent." to "…less than or equal to 4.6 weight percent",
- TS 5.6.1.c.2 is changed from "… Table 5.6-1 and Table 5.6-2." to "…Table 5.6-1, Table 5.6-2, and Table 5.6-3."
- TS 5.6.1.c.3 is changed from "… Table 5.6-1 and Table 5.6-2." to "…Table 5.6-1, Table 5.6-2, and Table 5.6-3",
- TS 5.6.1.d is changed from "…having a U-235 enrichment less than or equal to 4.5 weight percent…" to "…having a maximum planar average U-235 enrichment less than or equal to 4.6 weight percent…",
- FIGURE 5.6-1 the Allowed Special Arrangements Middle Figure – Above the Figure, "See Note 6" is added,
- FIGURE 5.6-1 in the NOTES, Note 1 is changed from "… Tables 5.6-1 and 5.6-2." to "…Table 5.6-1, Table 5.6-2, and Table 5.6-3",
- FIGURE 5.6-1 new NOTE 6 is added that states, "6. This arrangement is valid only for fresh assemblies with planar average enrichment ≤ 4.5 w/o.",
- FIGURE 5.6-2 in the NOTES, Note 1 is changed from "… Tables 5.6-1 and 5.6-2." to "…Table 5.6-1, Table 5.6-2, and Table 5.6-3",
- "TABLE 5.6-1 Minimum Burnup as a Function of Enrichment for Non-Blanketed Assemblies" is revised to add an "*" at the end of the title,
- TABLE 5.6-1, A footnote is added that states "* Operated at ≤ 2700 MWt",
- "TABLE 5.6-2 Minimum Burnup as a Function of Enrichment for Blanketed Assemblies" is revised to add an "*" at the end of the title,
- TABLE 5.6-2, A footnote is added that states "* Operated at ≤ 2700 MWt", and
- New "TABLE 5.6-3 Minimum Burnup as a Function of Enrichment for Blanketed Assemblies (Operated at ≤ 3020 MWt)" is added.

<u>Licensing Report:</u> Section 2.8.2, Nuclear Design; Section 2.8.6.1, New Fuel Storage; and Section 2.8.6.2, Spent Fuel Storage

Basis for the Change: Changes to Design Features - Fuel Storage - Criticality are related to new EPU fuel minimum burnup as a function of enrichment for blanketed assemblies, the storage of fuel with a slight increased U-235 in enrichment from 4.5 to 4.6 weight percent, and the balance of the changes are administrative.

Changes to TS 5.6.1.a.4, TS 5.6.1.a.6 and TS 5.6.1.c are administrative, as they do not change the Criteria, but provide the exact location within the section for the applicable criteria.

Changes to TS 5.6.1.c.2, TS 5.6.1.c.3, Note 1 on TS Figure 5.6-1, and Note 1 on TS Figure 5.6-2 are administrative as they reference the addition of the new TS Table 5.6-3 for the newer EPU fuel minimum burnup data for blanketed assemblies.

TS Figure 5.6-2 is changed to add Note 6 that prevents using one of the special arrangements for fresh fuel with the higher enrichment fuel (> 4.5 weight percent).

Change to TS 5.6.1.c.1 increases the average U-235 enrichment maximum from 4.5 to 4.6 weight percent for the loading of spent fuel storage racks. A new criticality analysis was conducted for the spent fuel storage racks and the cask pit storage racks using the EPU changes in enrichment to a maximum planar average value of 4.6 weight percent U-235 and the anticipated core operating characteristics for EPU conditions. These analyses were conducted to assure that the current licensing basis for Regions 1 and 2 loading patterns were conservative. Analyses were also performed for the loss of soluble boron in the spent fuel pool. The results are within the regulatory acceptance shutdown margin limits of K_{eff} less than 0.95 for the spent fuel storage racks K_{eff} below 1.00 for a loss of all soluble boron in the pool. Hence, the current loading patterns for the spent fuel pool and cask storage pit are within the current licensing basis.

Change to TS 5.6.1.d increases the average U-235 enrichment maximum from 4.5 to 4.6 weight percent for the loading of new fuel in the new fuel storage racks. Since EPU requires the enrichment of new fuel to increase to a maximum planar average of 4.6 weight percent U-235, a new bounding evaluation was performed assessing the storage of fresh fuel with the new enrichment. The results are within the regulatory acceptance shutdown margin limits of K_{eff} less than 0.95 for full water density and of K_{eff} less than 0.98 for optimum moderation conditions. Hence the current loading patterns for the new fuel storage racks are within the current licensing basis.

As discussed above, the EPU fuel will use a slightly increased enrichment. Burnup calculations for potential fuel types as a function of cooling, initial enrichment of EPU fuel and expected core operating conditions have been completed and are inserted as TS Table 5.6-3. TS Tables 5.6-1 and 5.6-2 are footnoted to identify that the data supports pre-EPU operation at or below an RTP of 2700 MWt.

32. TS 6.8.4, ADMINISTRATIVE CONTROLS - PROGRAMS

- TS 6.8.4.h. Containment Leakage Rate Testing Program is changed from “The peak calculated containment internal pressure for the design basis loss-of-coolant accident P_a, is 39.6 psig.” to “The peak calculated containment internal pressure for the design basis loss-of-coolant accident P_a, is 42.8 psig.”

Licensing Report: Section 2.6.1, Containment Functional Design

Basis for the Change: The revised containment pressure LCO of + 0.5 psig for initial containment pressure will limit the total pressure to less than 44.0 psig which is the design pressure and is consistent with the accident analyses The peak containment pressure obtained from the design basis LOCA to 42.8 psig. This change provides the value for the peak pressure following the design basis LOCA.

33. TS 6.9.11, ADMINISTRATIVE CONTROLS – CORE OPERATING LIMITS REPORT (COLR)

- TS 6.9.11.b The references for the COLR are revised to address the revised analyses for EPU. The following references are deleted and replaced with "DELETED".

  4 ANF-84-73(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events",

  6 EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs",

  8 XN-NF-82-49(P)(A), "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model",

  11 EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications", and

  15 ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors Analysis of Non-LOCA Chapter 15 Events"

- TS 6.9.11.b The following reference is being added:

  - 23. EMF-2103(P)(A) Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors" Framatone ANP, Inc., April 2003

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 a new reference has been included.

Licensing Basis Changes

1. Related License Amendment Requests (LARs)

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

2. Realistic Large Break Loss-of-Coolant Accident (RLBLOCA)

   The RLBLOCA analysis methodology has been applied to the LBLOCA analysis at EPU conditions. Attachment 5, Appendix C to this LAR provides the RLBLOCA summary report. This analysis methodology provides best-estimates of peak clad temperature and other parameters associated with LBLOCA analyses. This methodology has been reviewed and approved for use by the NRC. In performing the St. Lucie Unit 1 analyses, all limitations and restrictions have been met, including those specified by the NRC. The results show that all requirements of 10 CFR 50.46 have been met.

3. Control Room Dose Increase

   The dose analysis for the EPU indicates that the increase in control room dose for the LOCA, steam generator tube rupture (SGTR), locked rotor, fuel handling accident (FHA), CEA ejection, and gas decay tank rupture design basis accidents (DBAs) exceed the threshold for minimal increase under 10 CFR 50.59 and will require NRC review and approval. The dose analyses and results are summarized in LR Section 2.9.2, Radiological Consequences Analyses Using Alternative Source Team and Section 2.9.3, Radiological Consequences for Gas Decay Tank Ruptures. An increase in licensed power results in an increase in source term and, therefore, predicted dose is expected to increase.

   FPL has calculated doses for all of the DBAs required by NRC Regulatory Guide (RG) 1.183, Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants (Reference 11), and NUREG-0800 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP) Section 15.0.1 Radiological Consequences Analyses Using Alternative Source Terms (Reference 12). Doses were calculated for exclusion area boundary (EAB), low population zone (LPZ), and control room for each accident. The calculated doses for the LOCA, SGTR, locked rotor, FHA, CEA ejection, and gas decay tank rupture accidents exceeded the 10 percent minimal increase criteria. However, these doses are acceptable because they remain less that 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.

4. Plant Modifications

   The installation of the following modifications required for EPU is performed pursuant to 10 CFR 50.59.

   Approval of this LAR is required to operate these systems at EPU conditions.

   - Increase the safety injection tank nitrogen pressure from 200–250 psig to 230–280 psig.
   - Use of 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 Renewed Facility Operating License, Technical Specifications and current licensing basis, 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 1 Renewed Facility Operating License (FOL) DPR-67 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 1 current licensing basis (CLB). The changes to the FOL, TS, and CLB have been grouped and evaluated pursuant to 10 CFR 50.92.

A. Reactor Core Power Level

The FOL 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 St. Lucie Unit 1 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 RG 1.183, BTP 11-5, and 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. 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. Reactor Coolant Flow Rate and Departure From Nucleate Boiling (DNB) Parameters

The reactor core thermal margin safety limit for reactor coolant flow is increased from 365,000 gpm to 375,000 gpm and the reactor coolant system (RCS) cold leg temperature is increased from 549°F to 551°F. The DNB-related parameters (cold leg temperature, pressurizer pressure, and RCS total flow rate) are determined on a cycle-specific basis and are being relocated to the core operating limits report (COLR).

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

   The limits on 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 DNB-related parameters are not accident initiators.

The revised DNB-related parameters were used in the accident and transient analyses which demonstrated that the reactor core thermal margin safety limits are not exceeded. The proposed change in the measurement flow limit does not impact the actual RCS flow and has no direct impact on the reactor coolant pumps (RCPs) or any other equipment.

The specific limits for the DNB-related parameters are being relocated from the TS to the COLR. The relocation of the DNB-related parameters has no direct impact on safety-related SSCs and no impact on the probability or 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 DNB-related parameters limits are consistent with the safety analyses assumptions. 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.

   The relocation of the DNB-related parameters has no direct impact on safety-related SSCs and does not create the probability of a new or different kind of accident.

   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 accident analyses demonstrate that all safety limits and design basis limits are met assuming the change in minimum reactor coolant flow and RCS cold leg temperature. The other DNB-related parameters being relocated are unchanged for EPU.

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

C. Steam Generator Water Level Limiting Safety System Setting

The steam generator 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 ≥ 34.78%.

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 steam generator 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 steam generator 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 Systems

The ECCS-related systems - operating TS is being revised, such that each ECCS-related system which currently includes high pressure safety injection pump and low pressure safety injection pump, also includes a charging pump. The small break loss-of-coolant accident (SBLOCA) credits the use of the charging pumps for makeup to the reactor coolant system (RCS). The proposed change revises the TS Limiting Condition for Operation and Surveillance Requirements to ensure that a charging pump, as well as the required flow path, is available when required. The proposed change is consistent with the St. Lucie Unit 2 TS.

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

   The ECCS-related systems are used to mitigate the consequences of an accident, and are not accident initiators. Accordingly, the proposed changes to the ECCS-related systems do not increase the probability of an accident previously evaluated.

Dose consequences were evaluated using EPU conditions. The operation of the ECCS-related systems is unchanged as a result of the proposed change. ECCS-related systems will continue to function as designed and performance requirements will continue to be met. The change to the boration systems flow path TS will ensure that emergency core cooling is available as required. 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 single failures are introduced as a result of the proposed change. SSCs previously required for mitigation of a transient continue to be capable of fulfilling their design functions. The proposed change has no adverse affect on any safety-related system or component and does not change the performance or integrity of any safety-related system.

   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 design basis limits for the accident and transient analyses are met at EPU conditions. The ECCS-related systems will continue to met their design basis criteria.

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

E. Emergency Core Cooling Systems Pressure and Boron Concentration

The ECCS TS are being revised to change safety injection tank (SIT) nitrogen cover pressure from 200–250 psig to 230–280 psig and the minimum boron concentration in the SIT and refueling water tank (RWT) from ≥ 1720 ppm to ≥ 1900 ppm.

The boration control systems - shutdown margin, borated water sources, refueling operations, and special test exceptions TS are revised to address the increase in minimum 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 SIT component of the ECCS is qualified to operate at pressures greater than 280 psig. Accordingly, the increase in SIT pressure does not increase the probability of an accident previously evaluated.

   The proposed changes to the shutdown margin, boration water sources, refueling operations, and special test exceptions TS to increase in the minimum boron concentration for these TS have been evaluated. The proposed changes do not conflict with the ECCS TS. Components and systems will continue to function as designed and performance requirements for these will continue to be satisfied.

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 single failures are introduced as a result of the proposed change. 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 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 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 change does not result in a significant reduction in a margin of safety.

F. Reactor Coolant System Specific Activity

The $\overline{E}$ Average Disintegration Energy is being deleted and replaced with Dose Equivalent XE-133, consistent with TSTF-490, Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec. Accordingly, the gas storage tank TS is 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. The EPU accident analyses demonstrate that the resultant dose consequences remain within the applicable limits.

   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 in specific activity limits does not alter any physical part of the plant nor does it affect any plant operating parameter. The change does not create the potential for a new or different kind of accident previously calculated.

   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 proposed change revised the limits on noble gas radioactivity in the primary coolant. The proposed change is 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 change does not involve a significant reduction in a margin of safety.

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

The pressure-temperature (P/T) limit curves are being replaced with new curves that support operation to 54 effective full power years (EFPY). The revise P/T limit curves provide new temperature requirements for adjusting PORV setpoints to support LTOP and eliminate the need to isolate the high pressure safety injection pump headers when using the charging pumps for reactivity control in Modes 5 and 6.

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

   The proposed changes 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, Design Criteria 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 Regulatory Guide (RG) 1.99, Revision 2.

   The proposed changes will not result in physical changes to the SSCs of to event initiators or precursors. Changing the heatup and cooldown curves and the pressure relief setpoints to reflect 54 EFPY does not affect the ability to control the RCS at low temperatures such that the integrity of the reactor coolant pressure boundary would be compromised by violating the P/T limits.

   The proposed changes 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 changes 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 revised curves are based on a later edition of ASME Code Section XI that incorporates current industry standards for P/T curves. The revised curves are based on reactor vessel irradiation damage predictions using RG 1.99, Revision 2 methodology. 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 proposed P/T curves continue to maintain the safety margins of 10 CFR 50, Appendix G 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.

H. Containment System – Internal Pressure

The containment limit on positive pressure is being changed from + 2.4 psig to + 0.5 psig. In addition, the peak calculated containment pressure for the design basis LOCA (P_a) is increased to 42.8 psig.

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

   The containment and associated systems and components are used to mitigate the consequences of an accident, and are not accident initiators. Accordingly, the containment and associated systems and components do not increase the probability of an accident previously evaluated.

   The reduction in the normal operating maximum positive containment pressure ensures that the peak containment pressure following a LOCA does not exceed the containment design pressure and continues to satisfy the event acceptance criteria.

   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 proposed change to the containment internal pressure for the design basis LOCA. The proposed change in containment pressure is more restrictive than the current limit and has no adverse effect on any safety-related system and does 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 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.

I. Main Steam Safety Valves

The main steam safety valve (MSSV) as-found setpoint tolerance limits are being increased to account for setpoint drift. The MSSV lift setpoints 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 setpoint does not impact the probability of any accident. previously evaluated. The applicable accident analyses demonstrated acceptable results assuming the opening setpoint of the safety valves 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 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 safety valves was biased to the revised upper tolerance limits.

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

J. Condensate Storage Tank

The change in the minimum water volume requirement for the condensate storage tank (CST) accounts for the increase in water necessary for decay heat removal at EPU conditions, instrument uncertainty, and level above the outlet nozzle to prevent vortexing.

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

   The increase in the minimum CST water volume facilitates mitigation of accident consequences and does not initiate accidents. Additionally, the operation of the CST is unchanged for EPU. SSCs required to mitigate transients continue to be capable of performing their design functions. Acceptance criteria continue to be satisfied.

   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 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 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?

   Analyses demonstrate that the CST will continue to satisfy its functional safety and design criteria. The increase in CST minimum volume restores margin for accident and transient analyses.

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

K. Electrical Power Sources

The emergency diesel generator (EDG) minimum fuel oil volume requirement is being increased to 19,000 gpm. This change supports the use of ultra low sulfur fuel oil which has a lower heat rate (currently addressed by an administrative limit) and a small increase in the fuel oil requirements to support EDG loading for EPU.

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.

L. Electrical Power Sources

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 that 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 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.

M. Refueling Operations – Storage Pool Decay Time

This is a conforming change that addresses previously approved TS Amendment 190, issued by NRC letter dated April 28, 2004 (ML040440111). This TS is being deleted. FPL has installed single-failure proof cask cranes and previously removed the cask drop accident from the current licensing basis.

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

   The proposed elimination of this TS does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor does the change alter any assumptions or conditions in any plant accident 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 elimination of this TS will not affect the design function of any SSC. The change does not involve the addition or modification of equipment, not does it alter the design or operation of plant systems.

   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 proposed elimination of this TS does not alter the basis for any TS that is related to the establishment of or the maintenance of nuclear safety margin.

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

N. Design Features – Fuel Assemblies

The description of fuel assemblies is changed to provide a more accurate description and is consistent with the level of detail in NUREG-1432 and St. Lucie Unit 2 TS.

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

   These changes are administrative in nature and do not affect the operation of St. Lucie Unit 1. Components and systems will continue to function performance requirements for these systems will continue to be satisfied and no safety limits will be exceeded. The proposed administrative changes were not found to initiate any 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 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. No new safety-related equipment is being added or replaced as a result of the proposed administrative changes.

   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 are administrative in nature and do not affect any safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

O. Design Features – RCS Volume

RCS volume is not a measured or calculated parameter. NUREG-1432, Revision 3, Standard Technical Specifications Combustion Engineering Plants does not contain this TS. This TS is being deleted.

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

   The proposed elimination of the RCS Volume TS does not involve any physical changes to the plant. The elimination of this TS does not change to the configuration or method of operation of any plant equipment, nor does the change alter any assumptions or conditions in any plant accident 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 elimination of the RCS Volume TS will not affect the design function of any SSC. The elimination of this TS does not involve physical additions, or modifications to plant equipment, not does it alter the design or operation of plant systems.

   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 proposed elimination of the RCS Volume TS does not involve any physical changes to the plant, nor alter the basis for any TS that is related to the establishment of or the maintenance of nuclear safety margin.

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

P. Design Features – Fuel Storage

The design features TS are being revised to address the increase in fuel enrichment from a maximum planar average 4.5 weight percent U-235 to a maximum planar average 4.6 weight percent U-235, and a fuel burnup for operation at 3020 MWt.

Administrative changes are being incorporated that provide a more accurate location for criteria specified in this section, identified that a new table is being incorporated to address spent fuel that has been operated at ≤ 3020 MWt, and to add a footnote to the existing spent fuel tables to indicate that they had been operated at ≤ 2700 MWt.

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 fuel pool criticality analysis was conducted using the increased fuel enrichment and burnup. The analysis determined that the fuel pool loading patterns and a loss of soluble boron in the spent fuel pool event continue to satisfy the applicable acceptance criteria. An evaluation of the fresh fuel using the increase enrichment determined that storage of fresh fuel continued to satisfy the applicable acceptance criteria.

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

The administrative changes to the TS do not affect plant operation. 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 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.

   The administrative changes have no adverse effect on any safety-related systems and do 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 changes.

   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 change do not affect any safety analyses.

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

Q. Core Operating Limits Report 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 and listed in the COLR. The changes do not involve an physical changes to the plant and do not affect the operation of St. Lucie Unit 1. 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?

   These changes are administrative in nature and do not involve any physical changes to the plant, nor do they affect any safety analyses.

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

R. Realistic Large Break Loss-of-Coolant Accident (RLBLOCA) Methodology

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

   The RLBLOCA analysis methodology has been applied to the large break LOCA (LBLOCA) analysis at EPU conditions. The use of the RLBLOCA analysis methodology is not an accident initiator. Accordingly, the use of the RLBLOCA methodology does not increase the probability of an accident previously evaluated.

   The analysis methodology provides best-estimates of peak clad temperature and other parameters associated with LBLOCA analyses. This methodology has been reviewed and approved for use by the NRC. In performing the St. Lucie Unit 1 analyses, all limitations and restrictions have been met, including those specified by the NRC. The results show that all requirements of 10 CFR 50.46 have been met.

   Therefore, this change in methodology 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 changes involve the methodology used to calculate core response following a LBLOCA. There are no changes in hardware and consequently no new failure modes.

   Therefore, this change in methodology 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 RLBLOCA analysis results meet all the requirements of 10 CFR 50.46. The NRC has approved this methodology and all limitations and restrictions have been met in applying the methodology to St. Lucie Unit 1.

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

S. Control Room Dose Increase

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

   The determination of control room dose increase 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 change in increase in control room 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 1 FOL, 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 $\bar{E}$ 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. B. T. Moroney, Project Manager, NRC Letter to J. A. Stall, Senior Vice President (FPL), St. Lucie Units 1 and 2 – Issuance of Amendments Regarding the Relocation of Spent Fuel Pool Crane Technical Specification Requirements (TAC MB5667 and MB5668), April 28, 2004 (ML040440111) (Amendment 190).

11. NRC Regulatory Guide 1.183, Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants, July 2000.

12. NUREG-0800 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.