Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

Proprietary Information – Withhold From Public Disclosure Under 10 CFR 2.390
The balance of this letter may be considered non-proprietary upon removal of Attachment 2.

November 23, 2011

L-2011-493
10 CFR 50.90
10 CFR 2.390

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Re: St. Lucie Plant Unit 2
Docket No. 50-389
Renewed Facility Operating License No. NPF-16

Response to NRC Reactor Systems Branch and Nuclear Performance Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request

References:

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-021), "License Amendment Request for Extended Power Uprate," February 25, 2011, Accession No. ML110730116.

(2) Email from T. Orf (NRC) to C. Wasik (FPL), "St. Lucie 2 EPU draft RAIs – Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB)," September 6, 2011.

(3) Email from L. Abbott (FPL) to T. Orf (NRC), "St. Lucie 2 EPU draft RAIs – Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB) – Question Numbering," September 28, 2011.

By letter L-2011-021 dated February 25, 2011 [Reference 1], Florida Power & Light Company (FPL) requested to amend Renewed Facility Operating License No. NPF-16 and revise the St. Lucie Unit 2 Technical Specifications (TS). The proposed amendment will increase the unit's licensed core thermal power level from 2700 megawatts thermal (MWt) to 3020 MWt and revise the Renewed Facility Operating License and TS to support operation at this increased core thermal power level. This represents an approximate increase of 11.85% and is therefore considered an extended power uprate (EPU).

A001
NRR

In an email dated September 6, 2011 from NRC (T. Orf) to FPL (C. Wasik), Subject: St. Lucie 2 EPU draft RAIs – Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB), the NRC staff requested additional information regarding FPL's license amendment request (LAR) to implement the EPU. FPL email dated September 28, 2011 from FPL (L. Abbott) to NRC (T. Orf), Subject: St. Lucie 2 EPU draft RAIs – Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB) – Question Numbering, provided specific numbers (SXRB-1 through SRXB-102) for the questions included in the September 6, 2011 email. Attachments 1 and 2 to this letter provide the FPL responses to RAI questions SRXB-20 through SRXB-31 related to fuel, thermal-hydraulic and control rod drive mechanism design.

Attachment 1 contains the non-proprietary responses to RAI questions SRXB-20 through SRXB-31. Attachment 2 contains the proprietary responses to RAI questions SRXB-30 and SRXB-31.

Attachment 3 contains a copy of the Proprietary Information Affidavit. The purpose of this attachment is to withhold the proprietary information contained in the responses to SRXB-30 and SRXB-31 (Attachment 2) from public disclosure. The Affidavit signed by Westinghouse as the owner of the information sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of § 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390.

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the designated State of Florida official.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2011-021 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

Should you have any questions regarding this submittal, please contact Mr. Christopher Wasik, St. Lucie Extended Power Uprate LAR Project Manager, at 772-467-7138.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed on 23-November-2011

Very truly yours,

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachments (3)

cc: Mr. William Passetti, Florida Department of Health

Response to Reactor Systems Branch and Nuclear Performance Branch Request for Additional Information

The following information is provided by Florida Power & Light (FPL) in response to the U.S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support the Extended Power Uprate (EPU) License Amendment Request (LAR) for St. Lucie Nuclear Plant Unit 2 that was submitted to the NRC by FPL via letter (L-2011-021), February 25, 2011, Accession No. ML110730116.

In an email dated September 6, 2011 from NRC (T. Orf) to FPL (C. Wasik), Subject: St. Lucie 2 EPU draft RAIs – Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB), the NRC staff requested additional information regarding FPL's request to implement the EPU. FPL email dated September 28, 2011 from FPL (L. Abbott) to NRC (T. Orf), Subject: St. Lucie 2 EPU draft RAIs – Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB) – Question Numbering, provided specific numbers (SXRB-1 through SRXB-102) for the questions included in the September 6, 2011 email. The responses to RAI questions SRXB-20 through SRXB-31 are provided in Attachments 1 (non-proprietary) and 2 (proprietary). The remaining responses will be provided in separate submittals.

The responses to SRXB-30 and SRXB-31 contain information that is proprietary to Westinghouse Electric Company (Westinghouse). As such, the non-proprietary responses for these RAIs are provided below. The proprietary responses are provided in Attachment 2.

I. Fuel, Nuclear, Thermal-Hydraulic and CRD Mechanism Design (Sections 2.8.1 Through 2.8.4.1 of Attachment 5, (Licensing Report)

SRXB-20 (RAI 2.8.1.2.4-1)

It is stated in section 2.8.1.2.4-1 that "the NRC-approved FATES3B model is used to calculate fuel rod performance over the irradiation history. FATES3B iteratively calculates the interrelated effects of temperature, pressure, cladding elastic and plastic behavior, fission gas release, and fuel densification, and swelling as a function of time and linear power."

Describe in detail, how the model is used iteratively to calculate the interrelated effects of temperature, pressure, cladding elastic and plastic behavior, fission gas release, and fuel densification, and swelling as a function of time and linear power under EPU conditions.

Response

The Westinghouse FATES3B Fuel Evaluation Model was designed to calculate the steady-state fuel rod temperature distribution, gap conductance, fuel and clad dimensions, fuel rod internal pressure, and stored energy for nuclear fuel rods. Models are contained in FATES3B which describe the principle fuel rod behavioral phenomena, including thermal expansion, densification, fuel relocation, swelling, fission gas generation and release, cladding creep and growth, and cladding elastic deformation. Detailed descriptions of the models are contained in NRC approved methodology topical reports documented in References SRXB-20-1 through SRXB-20-7. The approved FATES3B methodology required no modifications for application to

EPU conditions. The following summarizes relevant aspects of the FATES3B calculation methods.

In FATES3B, the fuel pellet is assumed to be a right circular cylinder. It accommodates volumetric changes due to fuel thermal expansion, densification, relocation, and fission-induced swelling. The cladding conforms to the diameter dictated by the fuel pellet for computed conditions of pellet-clad contact. Prior to contact, the cladding is assumed to be free-standing. Thermal, elastic, creep, and growth components contribute to the hot cladding dimensions.

Discrete axial segments, for which independent radial thermal equilibrium calculations are performed, are utilized to model the fuel active length. The converged results for each segment are coupled to those of other segments through the assumption of complete and instantaneous mixing of the free gases within the fuel rod. The coupling permits integrated, whole rod predictions of fuel rod internal pressure.

A fuel pin history may be followed through a series of time increments during which the independent parameters in the various models are assumed constant. Fuel pin power history is based on 3-dimensional power peaking factors used in conjunction with previous and/or projected core behavior.

The FATES3B fuel performance code incorporates detailed models of fuel and cladding that were developed to describe gap closure, and to account for the effects of power history and axial power variation. An essential element of the FATES3B calculation method is accounting for the feedback effects of fuel and cladding temperatures, fuel gas release, fuel thermal expansion, densification, relocation, and fission-induced swelling and cladding thermal, elastic, creep, and growth components. Many of the behavior models, such as thermal expansion and fission gas release, are dependent on temperature. Fuel temperatures, however, are dependent on conditions in the fuel, especially conditions at the fuel-cladding interface. A fuel temperature solution is consequently iterated on at each fuel rod time point until a converged solution is obtained. The converged results for each discrete axial segment are coupled to the converged results at the other discrete axial segments through the complete and instantaneous mixing of the free gases within the fuel rod.

References

SRXB-20-1. CENPD-139-P-A, Fuel Evaluation Model, July 1974.

SRXB-20-2. CENPD-161(B)-P-A, Improvements to Fuel Evaluation Model, August 1989.

SRXB-20-3. CENPD-161(B)-P, Supplement 1-P-A, Improvements to Fuel Evaluation Model, January 1992.

SRXB-20-4. CENPD-275-P, Revision 1-P-A, C-E Methodology for Core Designs Containing Gadolinia-Urania Absorbers, May 1998.

SRXB-20-5. CEN-372-P-A, Fuel Rod Maximum Allowable Gas Pressure, May 1990.

SRXB-20-6. CENPD-275-P, Revision 1-P, Supplement 1-P-A, C-E Methodology for PWR Core Designs Containing Gadolinia-Urania Burnable Absorbers, April 1999.

SRXB-20-7. CENPD-404-P-A, Revision 0, Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs, November 2001.

SRXB-21 (RAI 2.8.1.2.4.2)

For Section 2.8.1.2.4.2 discussing input parameters, assumptions, and acceptance criteria,

(a) Explain what supporting analyses results, your statement that "since the different fuel configurations were explicitly evaluated, there is no impact due to having fuel with gadolinia and UO_2 composite fuel rods."

(b) Provide details of the distribution of Gadolinia (UO_2 + GD_2O_3) rods in the St. Lucie Unit 2 core for the upcoming EPU cycle, with respect to the number of Gadolinia rods and respective Gadolinium (Gd) enrichment.

(c) With degraded thermal conductivity, and lower melting point to the UO_2 + Gd_2O_3 mixture, describe what adjustments are made in the Gadolinia rods to prevent failure of the Gadolinia rod melting. Is there any restriction on linear heat generation rate (LHGR) limit for the Gadolinia rods during normal operation and anticipated operational occurrences (AOOs)?

(d) How is dependence of the thermal conductivity of the gadolinia rods on fuel burnup accounted for in the safety analyses?

(e) Discuss the impact of Gd content in Gadolinia rods (UO_2 + GD_2O_3) on fuel densification, swelling and fission gas release fuel rods.

(f) Discuss your procedure how the gadolinia rods are treated for the LOCA analysis preventing the Gadolinia rods becoming hot-rods.

Response

SRXB-21(a)

The FATES3B fuel performance methodology described in References SRXB-21-1 through SRXB-21-7 was used to perform fuel performance analyses for the reference analysis and will be used to perform the cycle specific fuel performance analyses at EPU conditions. Since each of the gadolinia fuel rod and UO_2 fuel rod designs were explicitly analyzed as described in and as required by Reference SRXB-21-6, a composite, bounding fuel rod design was not analyzed.

SRXB-21(b)

At this time, the EPU cycles have not been designed. The EPU cycles analyzed for the EPU licensing analysis are representative of what the actual design cycles will be. The methodology described in References SRXB-21-4 and SRXB-21-6 was used to evaluate the gadolinia fuel rod analyses for the reference analysis and will be used to perform the cycle specific gadolinia fuel rod performance analysis at EPU conditions. The number of gadolinia rods per fuel assembly for those fuel assemblies that contained gadolinia fuel in the reference analysis for the EPU varied from 4 to 20. The concentration of Gd_2O_3 in UO_2 in the gadolinia fuel rods analyzed in the reference analysis was either 4.0, 6.0, or 8.0 wt%. The location of the Gd_2O_3-UO_2 rods within the fuel assemblies that contained gadolinia fuel in the reference analysis is the same as that previously used in pre-EPU cycles. Similar gadolinia loading schemes are anticipated to be used in the cycle specific EPU reactor cycles. The cycle specific EPU gadolinia loading schemes will conform to the methodology approved by the NRC in References SRXB-21-4 and SRXB-21-6.

SRXB-21(c)

The FATES3B fuel performance code explicitly accounts for reduced fuel thermal conductivity and fuel melting temperatures with the addition of gadolinium burnable absorber as described in Reference SRXB-21-6. As described in Reference SRXB-21-6, the design of the core will ensure that the peaking factors in the gadolinia fuel are proportionally lower as required to ensure that fuel melting is precluded during normal operation and anticipated operational occurrences.

SRXB-21(d)

The effects of reduced thermal conductivity due to the addition of materials such as gadolinia and fuel burnup are accounted for differently. The fuel thermal conductivity with the addition of gadolinia to the UO_2 pellets is reduced from the UO_2 fuel thermal conductivity based upon the gadolinia weight percent as described in Reference SRXB-21-6. The conductivity dependence on the fuel burnup is compensated for through conservatisms inherent in the FATES3B code and its application methodology, including accounting for peaking factor reduction as the fuel rod accrues burnup.

SRXB-21(e)

As described in References SRXB-21-4 and SRXB-21-6, adding gadolinia to UO_2 fuel results in a slight change in as-fabricated fuel density. The densification and swelling of the gadolinia fuel is treated in the same manner as UO_2 fuel, accounting for the slight difference in initial fuel density, and will have a slight impact on fuel temperatures. As also described in References SRXB-21-4 and SRXB-21-6, and more importantly in terms of fuel temperatures, adding gadolinia to UO_2 fuel causes reduced fuel thermal conductivity, resulting in higher fuel temperatures for comparable gadolinia and UO_2 fuel rods at the same linear heat rate. The higher fuel temperatures in a gadolinia fuel rod compared to a UO_2 fuel rod at the same linear heat rate will generally result in higher rod internal pressures in the gadolinia fuel rod than in the UO_2 fuel rod due to increased fission gas release at the higher fuel temperatures in the gadolinia fuel rod. As the gadolinia loading increases, these relative affects also increase. To offset these negative effects in a gadolinia fuel rod relative to a UO_2 fuel rod, the allowable powers are reduced in gadolinia fuel rod design to obtain acceptable fuel performance results. These phenomena and methods are described in the NRC approved topical reports in References SRXB-21-4 and SRXB-21-6.

SRXB-21(f)

The enrichment in the gadolinia rods is set low enough, which results in these rods having lower powers, so that the gadolinia rods are not limiting from a loss of coolant accident (LOCA) perspective. This is verified on a cycle specific basis by controlling the enrichment of gadolinia rods so that the relative power in the gadolinia rods stays below specified limits which ensure that UO_2 rods remain limiting.

References

SRXB-21-1. CENPD-139-P-A, Fuel Evaluation Model, July 1974.

SRXB-21-2. CENPD-161(B)-P-A, Improvements to Fuel Evaluation Model, August 1989.

SRXB-21-3. CENPD-161(B)-P, Supplement 1-P-A, Improvements to Fuel Evaluation Model, January 1992.

SRXB-21-4. CENPD-275-P, Revision 1-P-A, C-E Methodology for Core Designs Containing Gadolinia-Urania Absorbers, May 1998.

SRXB-21-5. CEN-372-P-A, Fuel Rod Maximum Allowable Gas Pressure, May 1990.

SRXB-21-6. CENPD-275-P, Revision 1-P, Supplement 1-P-A, C-E Methodology for PWR Core Designs Containing Gadolinia-Urania Burnable Absorbers, April 1999.

SRXB-21-7. CENPD-404-P-A, Revision 0, Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs, November 2001.

<u>SRXB-22 (RAI 2.8.1.2.4.4)</u>

The licensee states in Section 2.8.1.2.4.4 that the "fuel performance evaluations have been completed for EPU transition and equilibrium cycles to demonstrate that the design criteria can be satisfied for all fuel types in the core under planned EPU operating conditions."

Pursuant to Criterion 10 (Reactor Design) and Criterion 35 (Emergency core cooling) of Appendix A to 10 CFR 50 and the requirements of ECCS Evaluation models, the licensee is expected to incorporate a methodology for calculating the highest cladding and fuel temperatures and thereby the highest calculated stored energy in the fuel during any condition of normal operation, including AOOs and for ECCS evaluation models. This methodology shall include the evaluation of thermal conductivity of the fuel as a function of burnup and temperature taking into consideration all of the effects that take place in the fuel during irradiation including but not limited to solid fission product buildup both in solution and as precipitates, porosity, and fission gas–bubble formation. This evaluation shall also include the effects of thermal conductivity on all fuel rod thermal-mechanical analyses (e.g. rod internal pressure) and inputs downstream safety analyses (e.g. LOCA stored energy).

(a) Describe, in detail, the licensee's methodology to evaluate fuel thermal conductivity as a function of burnup and temperature considering all of the effects that take in the fuel during the irradiation in the reactor core.

(b) For the LOCA analysis for the EPU fuel cycle, has the degradation of thermal conductivity due to all of the transformations that take place during the irradiation of the fuel in the reactor core been considered?

<u>Response</u>

<u>SRXB-22(a)</u>

The FATES3B fuel performance code described in References SRXB-22-1 through SRXB-22-7 is used to calculate fuel temperatures and rod internal pressures for the safety analysis and thermal-mechanical calculations. The FATES3B code was developed to conservatively bound the fuel temperature and fission gas release data measured, and consequently provide a conservative prediction of rod internal pressure. In addition to the conservatisms built into the models, the application methodology of the FATES3B code, especially through the use of bounding power histories, bounding axial power shapes, and accounting for peaking factor reduction with burnup compensates for the effects of thermal conductivity degradation for the safety and design applications.

SRXB-22(b)

The STRIKIN-II LOCA fuel rod temperature model (Reference SRXB-22-8) coincides with the rod temperature model of the FATES3B code. That is, the fuel rod temperature formulation used in STRIKIN-II and FATES3B is the same and key model parameters are directly transferred from FATES3B to STRIKIN-II. Thus, as noted above for the FATES3B evaluation, STRIKIN-II also conservatively bounds the fuel temperature and uses conservative predicted rod internal pressure information provided by FATES3B. In addition, the methodology used for loss of coolant accident (LOCA) calculations with STRIKIN-II biases conservatively key parameters, including parameters from the FATES3B – STRIKIN-II interface data file, like use of the end of life FATES3B fuel rod conductivity values and end of life fuel rod density for all times in life in STRIKIN-II. This tends to conservatively minimize the fuel rod conductivity used in STRIKIN-II and to maximize the fuel rod thermal heat capacity for each cycle. In addition, the 10 CFR 50 Appendix K LOCA methodology uses very conservative assumptions that introduce additional margin to compensate for the effects of thermal conductivity degradation for LOCA applications.

References

SRXB-22-1. CENPD-139-P-A, Fuel Evaluation Model, July 1974.

SRXB-22-2. CENPD-161(B)-P-A, Improvements to Fuel Evaluation Model, August 1989.

SRXB-22-3. CENPD-161(B)-P, Supplement 1-P-A, Improvements to Fuel Evaluation Model, January 1992.

SRXB-22-4. CENPD-275-P, Revision 1-P-A, C-E Methodology for Core Designs Containing Gadolinia-Urania Absorbers, May 1998.

SRXB-22-5. CEN-372-P-A, Fuel Rod Maximum Allowable Gas Pressure, May 1990.

SRXB-22-6. CENPD-275-P, Revision 1-P, Supplement 1-P-A, C-E Methodology for PWR Core Designs Containing Gadolinia-Urania Burnable Absorbers, April 1999.

SRXB-22-7. CENPD-404-P-A, Revision 0, Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs, November 2001.

SRXB-22-8. CENPD-135 P STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program, August, 1974.

SRXB-23 (RAI 2.8.1.2.4.3)

For Section 2.8.1.2.4.3 addressing a maximum rod internal pressure, explain the evaluation method and the results that predicted maximum rod internal pressure will not exceed the critical pressure limit at any time in life for anticipated operation and AOOs at EPU conditions.

Response

FATES3B (Reference SRXB-23-1) was used to calculate the rod internal pressure and corresponding critical pressure limit according to the NRC approved methodology described in References SRXB-23-2 and SRXB-23-3. Where appropriate, the approved gadolinia methodology of References SRXB-23-4 and SRXB-23-5 has been applied.

The critical pressure limit is the internal hot gas pressure at which the outward tensile creep rate of the cladding exceeds the fuel pellet radial growth rate due to fuel swelling, thus creating an increasing fuel-clad gap.

The hot internal pressure of a limiting rod was calculated at all times in life, under EPU conditions, taking into account anticipated operational occurrences (AOOs), using the FATES3B code. These calculations account for fission gas release, cladding outward creep, fuel pellet swelling, and other effects as discussed in the response to RAI SRXB-20 above. The resulting calculated pressures were verified to be below the critical pressure limit at all times in life.

As part of the reload safety evaluation process, the maximum allowable rod internal pressure calculated at all times in life for EPU conditions is verified every cycle to remain below the critical pressure limit during normal operation and AOOs.

References

SRXB-23-1. CENPD-161(B)-P, Supplement 1-P-A, Improvements to Fuel Evaluation Model, January 1992.

SRXB-23-2. CEN-372-P-A, Fuel Rod Maximum Allowable Gas Pressure, May 1990.

SRXB-23-3. CENPD-404-P-A, Revision 0, Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs, November 2001.

SRXB-23-4. CENPD-275-P, Revision 1-P-A, C-E Methodology for Core Designs Containing Gadolinia-Urania Absorbers, May 1998.

SRXB-23-5. CENPD-275-P, Revision 1-P, Supplement 1-P-A, C-E Methodology for PWR Core Designs Containing Gadolinia-Urania Burnable Absorbers, April 1999.

SRXB-24 (RAI 2.8.1.2.4.3-H)

For Section 2.8.1.2.4.3-H addressing cladding creep collapse, provide a summary of the 'cladding creep analysis' that was performed to verify that the pellet to cladding gap does not close under EPU conditions.

Response

As noted in EPU LAR Attachment 5, Section 2.8.1.2.4., a detailed cladding creep analysis was performed considering EPU conditions to demonstrate that cladding collapse does not occur in any fuel assembly at any time during the fuel operating life, which is considered out to a burnup of 65,000 MWD/MTU. This evaluation considered differential cladding pressure, cladding temperature, cladding flux, and cladding thinning due to oxidation, all as a function of time. The methodology for the cladding creep analysis is discussed in CENPD-404-P-A, Implementation of ZIRLO® Cladding Material in CE Nuclear Power Fuel Assembly Designs, November 2001, which was reviewed and approved by the NRC.

To summarize, the methodology involves the use of the CEPANFL code which calculates cladding ovality as a function of time until the rate of ovality increase becomes excessive, at which time the cladding is considered to collapse. The code uses three categories of information to perform this calculation, 1) cladding properties, 2) operating conditions, and 3) pellet column gap length. For predicting the minimum operating time prior to cladding collapse for a particular rod design, the beginning of life (BOL) cladding properties of mid-wall radius, wall thickness, and ovality are defined based on worst case drawing dimensions. In addition, oxide thickness as a function of time is defined. The time dependent operating conditions include the differential pressure across the cladding (based on a conservative minimum internal rod pressure history from the FATES3B code), cladding temperature, and fast neutron flux.

The cladding is assumed to have an initial ovality. Differential pressure is applied to the cladding, resulting in a stress distribution that causes the cladding to creep due to thermal and irradiation effects. At the end of each specified time step, CEPANFL applies the ZIRLO® creep correlation to the stress distribution to calculate the creep strain that has occurred during the time step. Because the cladding is oval, the stress distribution along the circumference is non-uniform, resulting in non-uniform creep strain, which causes the cladding to become more oval. This process is repeated at the end of each time step until the rate of ovality increase becomes excessive. In the plenum region of the fuel rod, where there is no support from the pellets, this is considered to be the collapse time. In the active region of the fuel, where a finite gap length between pellets is assumed, this time is multiplied by a correction factor that credits the support from the pellets at both ends of the gap and is a function of the gap length, and the resulting time is considered to be the collapse time. Note that in this methodology, the collapse time is not related to closure of the pellet-to-cladding gap, as suggested by the question, because cladding collapse has not occurred when the gap has closed. Rather, when the gap has closed, the pellet supports the cladding so that collapse cannot occur. Collapse can only occur in a span of cladding that is not supported by pellets, such as in the plenum region or in the active fuel region where there is a large gap between pellets.

The EPU analyses demonstrated that for the expected fuel operating life-time, no cladding creep collapse was predicted.

SRXB-25 (RAI 2.8.1.2.4.3)

For Section 2.8.1.2.3.4.3 regarding description of analyses and evaluations, provide detailed summary of the following analyses and results and show how the mechanical fuel design criteria are satisfied under EPU conditions at Saint Lucie Unit 2.

(a) Clad Stress

(b) Clad Strain

(c) Clad Oxidation and Hydriding

(d) Clad Fatigue

(e) Assembly Growth and Shoulder Gap

(f) Rod Axial Growth

(g) Swelling and Rupture

(h) Overheating of Fuel Pellets

(i) Fuel Rod Ballooning

Response

Clad Stress - SRXB-25(a)

As noted in EPU LAR Attachment 5, Section 2.8.1.2.4.3-C, a detailed stress analysis was performed considering EPU conditions to demonstrate that the cladding and end cap weld tensile and compressive stresses are less than their allowable limits at the applicable temperatures. This evaluation considered differential cladding pressures, as well as, axial loads and bending moments resulting from rod handling and seismic and loss of coolant accident (LOCA) events. Conservative cladding dimensions were used, and loss of cladding wall thickness due to oxidation was accounted for. The methodology for the fuel rod stresses is discussed in CENPD-404-P-A (Reference SRXB-25-7), which was reviewed and approved by the NRC.

The cladding stresses (including the end cap weld stresses) consider both normal operating and upset conditions. The primary tensile and compressive stresses in the cladding shall not exceed 66.66% and 100% of the minimum unirradiated yield strength, respectively, at the applicable temperatures for these conditions. The stresses are also examined for emergency and accident conditions. The stress limits for these conditions are included in CENPD-178-P (Reference SRXB-25-9). The rod internal pressures used to perform the stress analyses of the fuel rod designs account for power dependent and time dependent changes, such as the fuel rod void volume, fission gas release and gas temperature, cladding creep and thermal expansion. The rod external pressures are consistent with the event being analyzed and are biased in the conservative direction (maximum for compressive stresses, minimum for tensile stresses). The maximum tensile and compressive stresses were calculated for fuel handling and storage, reactor servicing, beginning of life (BOL) rod withdrawal, power operation and reactor trip, heatup and cooldown, minor fuel handling accident, operating basis earthquake (OBE), design basis earthquake (DBE), loss of coolant accident (LOCA), and combined DBE + LOCA.

The maximum tensile stresses for the different events analyzed demonstrated that for the EPU, there was at least 10% margin to the allowable tensile stress limits. For the maximum compressive stresses, it was demonstrated that for the EPU there was at least 90% margin to the allowable compressive stress limits.

Clad Strain - SRXB-25(b)

As noted in EPU LAR Attachment 5, Section 2.8.1.2.4.3-D, the design limit for cladding strain is that the total plastic tensile creep strain due to uniform cladding creep and uniform cylindrical fuel pellet expansion, due to swelling and thermal expansion is less than 1% from the unirradiated condition, and that the total tensile strain due to uniform cylindrical pellet expansion during a transient is less than 1% from the pre-transient value. The methodology for the fuel rod cladding strain is discussed in CENPD-404-P-A (Reference SRXB-25-7).

The first part of the strain limit concerns the total plastic strain incurred as a result of cladding creep and cladding yielding during long term normal operation and short transient conditions. Cladding creep strain and plastic strain due to cladding yielding are driven by the stress in the cladding that results from differential pressure and interference with the fuel pellets. The methodology used to evaluate the strain accounts for power dependent and time dependent parameters, including differential pressure across the cladding (based on a conservative maximum internal rod pressure history from FATES3B), cladding temperature, pellet diameter, and clad diameter. To ensure that the calculations are conservative, the calculation considers peak local burnups that are based on a rod average burnup of 65,000 MWD/MTU.

To determine the permanent strain resulting from normal operation, differential pressure is applied to the cladding, resulting in a stress distribution that causes the cladding to creep due to thermal and irradiation effects. At the end of each specified time step, the ZIRLO® creep correlation is applied to the stress distribution to calculate the creep strain that has occurred during the time step, and the cladding diameter is adjusted to include the creep strain. At the same time, the change in pellet diameter due to thermal and irradiation effects is calculated, and the pellet diameter is adjusted accordingly. The new pellet diameter is compared to the new cladding diameter, and if there is interference between the two, the interference strain is compared to the yield strain to determine if yielding has occurred. (The cladding is assumed to conform to the predicted diameter of the pellet during periods of contact, i.e., compression of the pellet is conservatively ignored). Any strain that is higher than the yield strain is considered to be permanent strain, and the cladding diameter is adjusted accordingly. This process is repeated at the end of each time step, and the resulting cladding diameter is compared to the BOL diameter to determine the amount of permanent strain that has occurred. Early in life, the creep strain is typically compressive because the rod external pressure is greater than the rod internal pressure, and the pellet has not swelled enough to contact the cladding. As time goes by, the cladding strain reverses direction as the rod internal pressure increases and the pellet expands. In the case of the EPU, the reversal by end of life (EOL) is not sufficient to surpass the early compressive strain, and the final permanent strain due to normal operation remains slightly compressive. To satisfy the criterion, the normal operating plastic strain combined with the transient plastic strain must be less than 1% tensile. As discussed below, the total (plastic plus elastic) transient strain is less than 1%, so the transient plastic strain must be less than 1%, and when added to the normal operating plastic strain, which is compressive, the total plastic strain is less than 1%, and the criterion is satisfied.

Compliance with the transient induced strain limit is demonstrated by showing that the maximum pellet linear heat generation rate (LHR) during any anticipated operational occurrence (AOO) is less than a limiting curve of maximum LHR versus initial LHR, where the limiting curve is based on the calculated change in the LHR which results in 1.0% cladding strain. The limiting curve is generated in this calculation using the FATES3B code to determine fuel pellet outer diameter (OD) as a function of the LHR, and conservatively assuming that the pellet is infinitely rigid, such that the cladding offers no resistance to thermal expansion of the pellet, and immediately prior to the transient the pellet is in contact with the cladding.

Given the above for EPU, it was demonstrated that the total plastic tensile creep strain due to uniform cladding creep and uniform cylindrical fuel pellet expansion due to swelling and thermal expansion is less than 1% from the unirradiated condition, and that the total tensile strain due to uniform cylindrical pellet expansion during a transient is less than 1% from the pre-transient value.

Clad Oxidation and Hydriding SRXB-25(c)

The methodology and model described in the NRC approved CENPD-404-P-A (Reference SRXB-25-7) is used to confirm that the maximum best estimate fuel rod oxide thickness at any location along the fuel rod, and at any time in life, is less than the NRC approved criterion of 100 microns. The maximum best estimate fuel rod oxide thickness predicted in the EPU reference analyses is less than the criterion of 100 microns. And, the best estimate fuel rod oxide thickness will be evaluated in cycle specific EPU analyses and confirmed to be less than the criterion of 100 microns.

The NRC safety evaluation report (SER) for CENPD-404-P-A (Reference SRXB-25-7) also required that until data is available demonstrating the performance of ZIRLO® cladding in Combustion Engineering Nuclear Power (CENP) designed plants, the fuel duty will be limited for each CENP designed plant with some provision for adequate margin to account for variations in core design. The details of this condition are addressed on a plant specific basis during the approval to use ZIRLO® in a specific plant. Reference SRXB-25-8 provided approval for the modified Fuel Duty Index (mFDI) limits that are required for ZIRLO® fuel rod cladding applications. The fuel duty will be limited to a baseline mFDI of 600 with a provision for adequate margin to account for variations in core design. Specifically, the mFDI of each ZIRLO® clad fuel pin is restricted to 110 percent of the baseline mFDI of 600. For a fraction of the fuel pins in a limited number of assemblies (8), the mFDI of the ZIRLO® clad fuel pins is restricted to 120 percent of the baseline mFDI of 600. The methodology and model described in Reference SRXB-25-7 is used to confirm that the maximum mFDI for any fuel rod and at any time in life is less than the NRC approved criterion. The maximum mFDI predicted in the EPU reference analyses is less than the criterion. The maximum mFDI will be evaluated in cycle specific EPU analyses and confirmed to be less than the criterion.

Clad Fatigue – SRXB-25(d)

For the number and type of transients which occur during normal operation, the EOL cumulative fatigue damage factor in the cladding and in the end-cap welds must be less than 0.8. To support the EPU, a fuel rod fatigue evaluation, which considered the differential cladding pressures and cladding temperatures associated with the EPU, cladding creep, and pellet swelling, was performed which demonstrated that the above design requirement was satisfied. The methodology for the fuel rod cladding strain is discussed in CENPD-404-P-A (Reference SRXB-25-7).

Fatigue damage to the fuel rod cladding can be induced by reactor trips, startups/shutdowns, and power cycling. Each of these events results in cyclic changes in clad strain due to power changes that cause the differential pressure across the rod to vary and the pellet to come into and out of contact with the cladding. With respect to determining changes in cladding and pellet diameters, the methodology used to evaluate the fatigue damage from power cycling is the same as described above for the strain evaluation during normal operation, except that some of the parameters are biased in the opposite direction to provide results that are conservative for fatigue analysis. The power is conservatively assumed to vary between 10% and 100% on a daily basis, where 100% power is conservatively assumed to be the LOCA limit, and for each day, the maximum strain range resulting from the changes in power is determined. The allowable number of cycles for that strain range is determined based on the fatigue design curve, and the reciprocal of that number is the fatigue damage factor for that day. This process is repeated for each day of operation, where the number of days is the number required to achieve the peak local burnup, which is based on a rod average burnup of 65,000 MWD/MTU. The fatigue damage factors from all the days of operation are totaled to give the cumulative damage factor of 0.68 due to power cycling.

The methodology is similar for determining the damage associated with reactor trips, and startups/shutdowns. Sixty reactor trips between 100% power and hot standby and fifty startups/shutdowns between 100% power and room temperature conditions are considered. The resulting cumulative damage factors are 0.0075 and 0.0063, respectively.

The total cumulative fatigue damage factor from power cycling, reactor trips and startups/shutdowns is 0.69, which is below the limit of 0.8. Given the above for the EPU, it was demonstrated that the fuel rod fatigue damage factor criterion was satisfied.

Assembly Growth and Shoulder Gap – SRXB-25(e)

The axial length between the end fittings must be sufficient to accommodate the differential thermal expansion and irradiation-induced differential growth between the fuel rods and the guide tubes, such that it can be shown with a 95 percent confidence level that no interference exists. The EPU conditions increase the shoulder gap margin relative to the current operating conditions since the slight increase in the uplift forces on the spacer grids associated with the EPU increases the guide thimble growth slightly, thus increasing the shoulder gap.

The shoulder gap is affected by the fuel assembly growth and the fuel rod growth. Prior analyses of the same fuel assembly design, but with non-EPU conditions have demonstrated adequate shoulder gap margin. Relative to these analyses, the only change that affects fuel assembly growth is a slight increase in uplift forces on the spacer grids due to the slight increase in coolant temperature associated with EPU. The slight increase in uplift forces results in a slight increase in fuel assembly growth, which increases shoulder gap. Fuel rod growth is not affected because it is solely a function of fuel rod fluence, and the fuel rod fluence limit is the same for EPU as for non-EPU conditions. Therefore, it was concluded that the shoulder gap will be slightly greater for EPU conditions than for non-EPU conditions, and since the shoulder gap is acceptable for non-EPU conditions, it will also be acceptable for EPU conditions.

Rod Axial Growth – RSXB-25(f)

There is no separate mechanical design criterion for the fuel rod growth. Rather, the effect of the fuel rod growth is captured in the shoulder gap criterion, which is discussed above. It is important to note that there is no change in the maximum predicted fuel rod growth for EPU, because the fuel rod growth is solely a function of fluence, and the bounding fluence used for EPU is the same as for pre-EPU conditions. Finally, the same growth correlation is applicable to both ZIRLO® and Optimized ZIRLO®.

Swelling and Rupture – SRXB-25(g)

NUREG-0800, Standard Review Plan (SRP) 4.2 describes that swelling and rupture of the cladding, resulting from the cladding temperature distribution and the differential pressure across the cladding, should be accounted for during postulated accidents. The following description addresses swelling and rupture evaluations with respect to steady-state operation.

Analysis of fuel rod swelling and rupture are dependent on rod internal pressure, cladding creep, and cladding stresses. The swelling and rupture phenomena are not analyzed explicitly during normal operation and AOO conditions because this criterion is protected against by ensuring that 1) fuel to cladding gap reopening does not occur, 2) the cladding strain criterion is met, and 3) the cladding stress criteria is met. The calculations to protect against gap reopening were performed under EPU conditions in the reference analyses using the FATES3B code to ensure that maximum rod internal pressures are maintained below the critical gap reopening pressures. The calculated steady-state rod internal pressures are well below pressures required to cause rupture. The maximum rod internal pressure calculated at all times in life for EPU conditions in the cycle specific analyses is verified every cycle to remain below the critical pressure limit. The rod internal pressure analysis was performed using licensed methodology found in References SRXB-25-5, SRXB-25-6, and SRXB-25-7. Additionally, the cladding strain and stress analyses were performed under EPU conditions using licensed methodology as described earlier in this response.

<u>Overheating of Fuel Pellets – SRXB-25(h)</u>

The NRC approved FATES3B code and methodology of References SRXB-25-1 through SRXB-25-7, including the NRC approved methodology for gadolinia fuel rods (References SRXB-25-4 and SRXB-25-6) when appropriate, is used to calculate the maximum fuel rod temperatures during normal operation and AOOs. The maximum calculated temperatures are shown to be less than the fuel melting temperatures at any time in life. The fuel melting temperature for gadolinia fuel is less than the fuel melting temperature for a comparable UO_2 fuel rod, as described in the NRC approved Topical Report of Reference SRXB-25-6. The maximum fuel rod temperatures calculated for gadolinia and UO_2 fuel rods at all times in life for EPU conditions in the reference analyses remain below the fuel melting temperatures during normal operation and AOOs. The maximum fuel rod temperatures calculated for gadolinia and UO_2 fuel rods at all times in life for EPU conditions in the cycle specific analyses are verified every cycle to remain below the fuel melting temperatures during normal operation and AOOs.

<u>Fuel Rod Ballooning – SRXB-25(i)</u>

SRP 4.2 describes that burst strain and flow blockage caused by ballooning of the cladding should be accounted for to ensure that the fuel maintains a coolable geometry during postulated accidents. The following explanation addresses fuel rod ballooning evaluations with respect to steady-state operation.

Fuel rod ballooning is expected to be driven by runaway feedback effects caused by an opening fuel rod cladding gap that elevate the rod internal pressure causing outward cladding creep that further increases the fuel rod cladding gap. During steady-state operation and AOOs, fuel rod ballooning is protected against by ensuring that fuel to cladding gap reopening does not occur due to elevated rod internal pressures. Since the gap does not reopen under steady-state EPU conditions, there will not be elevated temperatures and elevated fission gas release that could cause a runaway effect raising rod internal pressure, and ballooning will not occur.

The maximum allowable rod internal pressure calculated at all times in life for EPU conditions in the reference analyses, remain below the critical pressure limit during normal operation and AOOs. The maximum rod internal pressure calculated at all times in life for EPU conditions in the cycle specific analyses is verified every cycle to remain below the critical pressure limit during normal operation and AOOs.

References

SRXB-25-1. CENPD-139-P-A, Fuel Evaluation Model, July 1974.

SRXB-25-2. CENPD-161(B)-P-A, Improvements to Fuel Evaluation Model, August 1989.

SRXB-25-3. CENPD-161(B)-P, Supplement 1-P-A, Improvements to Fuel Evaluation Model, January 1992.

SRXB-25-4. CENPD-275-P, Revision 1-P-A, C-E Methodology for Core Designs Containing Gadolinia-Urania Absorbers, May 1998.

SRXB-25-5. CEN-372-P-A, Fuel Rod Maximum Allowable Gas Pressure, May 1990.

SRXB-25-6. CENPD-275-P, Revision 1-P, Supplement 1-P-A, C-E Methodology for PWR Core Designs Containing Gadolinia-Urania Burnable Absorbers, April 1999.

SRXB-25-7. CENPD-404-P-A, Revision 0, Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs, November 2001.

SRXB-25-8. Letter from B. T. Moroney (NRC) to J. A. Stall (Florida Power & Light Company), St. Lucie Plant, Unit 2 – Issuance of Amendment Regarding Change in Reload Methodology and Increase in Steam Generator Tube Plugging Limit (TAC No. MC1566), January 31, 2005.

SRXB-25-9. CENP-178-P, Rev. 1-P, Structural Analysis of Fuel Assemblies for Seismic and LOCA Loading, August 1981

SRXB-26 (RAI 2.8.2.2.2)

Describe the analysis procedure used to ensure that the shutdown margin is within the TS limit throughout the transition and equilibrium cycles of EPU operation. Specifically, address how the eigenvalue biases and uncertainties are determined and accounted for during the transition cycles.

Response

The methodology used for all nuclear design calculations is contained in the PHOENIX and ANC codes, which are described in Reference 2, listed in EPU LAR Attachment 5, Section 2.8.2.4. These are used in conjunction with the Reload Safety Analysis Checklist (RSAC) process, described in Reference 1, listed in EPU LAR Attachment 5, Section 2.8.2.4. Details of the analysis procedure used to compute shutdown margin follows.

A search for the most adverse single stuck rod was performed separately at beginning of cycle (BOC) and at end of cycle (EOC), and then used in the calculation of net rod worth.

The calculational sequence assumes the following biases and uncertainties:

1) Maximum insertion of the lead bank allowed at full power. worth
2) Axial power shape skewed towards the top of the core. This was accomplished by adjusting the axial xenon distribution. worth
3) Net rod worth uncertainty of 10%, decreasing its absolute value. value
4) An allowance of 50 pcm for any subcooled voids that might be present at the full power condition, and which would then collapse after the scram. scram

5) The initial value of core inlet temperature is assumed to be above the nominal inlet temperature plus uncertainties. This increases the power defect going from full power to zero power and thus, reduces the computed shutdown margin. margin.

Transition cycles are treated the same as the equilibrium cycles since the uncertainties as determined following the NRC approved methodology for reload safety evaluations are not affected by transition or equilibrium core aspects. The nuclear design methodology is approved for a wide range of core conditions, from cold shutdown through severe accidents. It has been used successfully in core with various styles of fuel management, from out-in through low leakage, and for a wide range of enrichments and poison loadings. In comparison to these ranges, the difference between equilibrium and transition cycles is minor.

Additionally, as part of the RSAC process, cycle specific analyses are performed to verify that the shutdown margin is within Technical Specification limits.

SRXB-27 (RAI 2.8.2.2.3)

For Section 2.8.2.2.3 regarding description of analyses and evaluations, discuss the validity and applicability of Reference CEN-386-P-A for the discharge burnups that are anticipated for the EPU cycles at St. Lucie Unit 2.

Response

CEN-386-P-A was approved by the NRC for 1-pin burnup values up to 60 GWD/T. All discharge burnups for both transition and equilibrium cycles examined for the EPU had values under this limit. Additionally, the Reload Safety Analysis Checklist (RSAC), used jointly by Westinghouse and by FPL, require this quantity to be checked every cycle.

SRXB-28 (RAI 2.8.2.2.3)

Describe in detail, how the nuclear design analysis for the EPU cores at St. Lucie Unit 2 employed standard analytical models and methods as of References 1, 2, and 3 listed in Section 2.8.2.4.

Response

The same nuclear design procedures have been used for the EPU calculations as are used for pre-EPU reload calculations. These were created and are maintained by Westinghouse, and are used jointly by Westinghouse and FPL for nuclear design analysis of every cycle. These procedures specify the use of PHOENIX-P and ANC for nuclear design calculations (Reference 2 listed in EPU LAR Attachment 5, Section 2.8.2.4). The procedures used for the physics calculations follow the Reload Safety Analysis Checklist (RSAC) process, consistent with the methodology of WCAP-9273 (Reference 1 listed in EPU LAR Attachment 5, Section 2.8.2.4).

SRXB-29 (RAI 2.8.3.1)

For Section 2.8.3.1 addressing thermal and hydraulic design-regulatory evaluation, provide justification for compliance with GDC 12 for suppression of reactor power oscillations. GDC Criterion 12 requires that the reactor core and associated coolant, control, and protection systems shall be designed to ensure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Response

As described in the EPU LAR Attachment 5, the effect of the negative power coefficient of reactivity, along with the coolant temperature maintained by the control element assemblies (CEAs) and the soluble boron provide fundamental mode stability. Therefore, power oscillations will not occur. The power distribution oscillations are also detected by neutron flux detectors and suppressed by CEAs.

Updated Final Safety Analysis Report (UFSAR) Section 4.4.1 states that the principal thermal-hydraulic design (THD) basis is the avoidance of thermally or hydraulically induced fuel damage during steady state operation and during anticipated operational occurrences (AOOs). The UFSAR also states that operating conditions do not lead to flow instabilities. THD utilizes safety analysis fuel design limits (SAFDLs) to ensure that the core power distribution does not result in fuel centerline melting or departure from nucleate boiling ratios (DNBRs) less than a limiting value. THD ensures that the SAFDLs are not violated by determining a minimum DNBR, such that there is a 95 percent probability with a 95 percent confidence level that departure from nucleate boiling (DNB) does not occur on the limiting fuel rod during either steady state operation or AOOs. DNB analyses were performed at EPU conditions to verify that their respective DNBR limits were met. The reactor protective system and the Technical Specifications also ensure that the SAFDLs are not exceeded by enforcing limiting conditions for operation (LCOs). These LCOs are evaluated such that the initial conditions assumed in the analysis of AOOs and postulated accidents are conservative with respect to allowed reactor conditions. Nuclear design limits on power distributions are related to these LCOs, which in turn determine inputs to THD analyses.

SRXB-30 (RAI 2.8.3.2.2.2)

For Section 2.8.3.2.2.2 addressing DNB methodology, provide detailed calculations showing the implementation of RTDP methodology and show how the various uncertainties listed in Section 2.8.3.2.2.2 are statistically combined to obtain the overall DNB uncertainty factors.

(a) Table 2.8.3-5 lists RTDP margin summary. Explain how the design limit DNBR is conservatively increased to provide DNBR margin to offset the effect of rod bow and any other DNB penalties that may occur.

(b) Explain how the rod bow DNBR penalty in Table 2.8.3-5 is calculated.

Response

In 2005, FPL transitioned St. Lucie Unit 2 to the WCAP-9272 Reload Methodology, which included the design limit (DL) departure from nucleate boiling ratio (DNBR) calculations using the revised thermal design procedure (RTDP) methodology detailed in WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989. The RTDP methodology remains unchanged in the EPU analysis.

According to the NRC-approved RTDP methodology issued in WCAP-11397-P-A, the following procedure is utilized to statistically combine the various uncertainties:

1) Nominal values are determined for the parameters listed in EPU LAR Attachment 5, Section 2.8.3.2.2.2.
2) A DNBR sensitivity analysis is performed by perturbing each nominal parameter by its uncertainty over a wide range of conditions and various axial power shapes. The sensitivities are calculated using the following equation:

   $$[\qquad]^{a,c}$$

   Where: s_i is the sensitivity of DNBR to parameter y_i
   1 and 2 denote different values of the parameter y_i

3) Standard deviations for each design parameter are calculated according to their probability distribution.
4) A coefficient of variation is determined with the following equation using the limiting sensitivity for each parameter:

   $$[\qquad]^{a,c}$$

5) The RTDP design limit (DLR) is calculated from the following equation:

   $$[\qquad]^{a,c}$$

(a) The DL DNBR is increased to a safety analysis limit (SAL) DNBR by a plant specific margin to offset any rod bow or other departure from nucleate boiling (DNB) penalties incurred for the reference analysis. The SAL DNBR is selected to provide sufficient margin to offset any anticipated DNB penalties, as well as retaining some additional margin for future cycle-specific DNB penalties which may be required. All RTDP DNB analyses are performed to the SAL DNBR, in order to preserve the retained margin. The margin is evaluated each cycle to ensure that the SAL DNBR provides sufficient margin. Cycle specific analyses resulting in a minimum calculated DNBR less than the SAL DNBR have an associated penalty and are tracked in a cycle specific DNBR margin summary table, along with the available margin. Analyses that result in a minimum calculated DNBR greater than the SAL DNBR are not credited margin.

(b) St. Lucie Unit 2 contains 16x16 Standard Combustion Engineering (CE)-type fuel and therefore, utilizes the NRC-approved CE-nuclear steam supply system (NSSS) rod bow penalty methodology issued in CENPD-225-P-A, Fuel & Poison Bowing, June 1983. The NRC approved the extrapolation of the channel closure data for 14x14 fuel for applicability to 16x16 fuel through an (L^2/I) dependence in CEN-289-P-A, Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2, December 1984. Using this approved methodology, the maximum rod bow penalty determined for St. Lucie Unit 2 was [ ]^{a,c}% at a burnup of [ ]^{a,c} MWD/MTU.

SRXB-31 (RAI 2.8.3)

Table 2.8.3-1 lists the thermal-hydraulic design parameters for comparison. (a) Realizing the fact that there is a 11.6% increase in the core flow rate, explain why there is a relatively large increase (41%) in pressure drop across the core. (b) Provide the evaluations and calculations to support this large change in pressure drop.

Response

EPU LAR Attachment 5, Table 2.8.3-1 inadvertently reported pressure drops that were based on different axial height boundaries for the pre-EPU and EPU conditions. The EPU pressure drop reported in the table includes the pressure drop across core support plate, while the pre-EPU value does not. When considering an initial axial height boundary that includes the core support plate, the pre-EPU pressure drop is [ ]^{a,c} psi and the EPU pressure drop is [ ]^{a,c} psi. The increase in pressure drop across the core is approximately [ ]^{a,c}%. The pressure drop due to grid losses would increase by the square of the flow increase percentage, or approximately [ ]^{a,c}%. However, the decrease in the friction loss coefficient due to the higher flow and the decrease in density due to the higher temperature offset a portion of the predicted [ ]^{a,c}% increase in pressure drop. Therefore, the [ ]^{a,c} % increase in pressure drop is consistent with the EPU parameter changes.

ATTACHMENT 3

Response to NRC Reactor Systems Branch and Nuclear Performance Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request

Westinghouse Electric Company Affidavit for Withhold Proprietary Information from Public Disclosure

This coversheet plus 7 pages

Westinghouse

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CAW-11-3309

November 10, 2011

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Responses to Thermal/Hydraulic Requests for Additional Information for the St. Lucie Unit 2 Extended Power Uprate License Amendment Request." (Proprietary)

References:

1. NRC E-Mail, T. Orf (NRC) to C. Wasik (FPL), "St. Lucie 2 EPU - Draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB)," September 6, 2011, 12:19 PM.

The proprietary information for which withholding is being requested is that included in the responses to the Requests for Additional Information (RAIs) designated as "30. RAI 2.8.3.2.2.2" and "31. RAI 2.8.3" transmitted by Reference 1, and further identified in Affidavit CAW-11-3309 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Florida Power and Light.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-11-3309, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

for J. A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

STATE OF CONNECTICUT:

ss Windsor Locks

COUNTY OF HARTFORD:

Before me, the undersigned authority, personally appeared C. M. Molnar, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

C. M. Molnar, Senior Engineer
Regulatory Compliance

Sworn to and subscribed before me
this 10th day of November 2011

Notary Public

Subscribed and Sworn to before me, a Notary Public, in and for County of Hartford and State of Connecticut. this 10 day of November, 2011.

JOAN GRAY
Notary Public
My Commission Expires January 31, 2012

(1) I am Senior Engineer, Regulatory Compliance, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in responses to Requests for Additional Information (RAIs) "30. RAI 2.8.3.2.2.2" and "31. RAI 2.8.3", for submittal to the Commission, being transmitted by Florida Power and Light letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The RAIs identified above are included in NRC E-Mail, T. Orf (NRC) to C. Wasik (FPL), "St. Lucie 2 EPU - Draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB)," September 6, 2011, 12:19 PM. The proprietary information as submitted by Westinghouse is that which supports the St. Lucie Unit 2 Extended Power Uprate (EPU) License Amendment Request (LAR), and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Support the St. Lucie Unit 2 EPU LAR by justifying the DNB methodology employed in EPU analyses and supporting the validity of the calculated core pressure drop under EPU conditions.

Further this information has substantial commercial value as follows:

(a) The information reveals aspects of Westinghouse DNB methodology and provides thermal-hydraulic data that could facilitate competitors' future analyses.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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