Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

June, 18 2012
3F0612-05

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 – Response to Request for Additional Information to Support NRC Nuclear Performance and Code Review Branch (SNPB) Technical Review of the CR-3 Extended Power Uprate LAR (TAC No. ME6527)

References:
1. CR-3 to NRC letter dated June 15, 2011, "Crystal River Unit 3 – License Amendment Request #309, Revision 0, Extended Power Uprate" (ADAMS Accession No. ML112070659)
2. Email from S. Lingam (NRC) to D. Westcott (CR-3) dated March 21, 2012, "Crystal River, Unit 3 EPU LAR - SNPB Draft RAIs (ME6527)"
3. NRC to CR-3 letter dated May 4, 2012, "Crystal River Unit 3 Nuclear Generating Plant – Request For Additional Information For Extended Power Uprate License Amendment Request (TAC No. ME6527)" (ADAMS Accession No. ML120830038)

Dear Sir:

By letter dated June 15, 2011, Florida Power Corporation, doing business as Progress Energy Florida, Inc., requested a license amendment to increase the rated thermal power level of Crystal River Unit 3 (CR-3) from 2609 megawatts (MWt) to 3014 MWt (Reference 1). On March 21, 2012, via electronic mail, the NRC provided a draft request for additional information (RAI) to support the SNPB technical review of the CR-3 Extended Power Uprate (EPU) License Amendment Request (LAR) (Reference 2). By teleconference on May 1, 2012, CR-3 discussed the draft RAI with the NRC to confirm an understanding of the information being requested. On May 4, 2012, the NRC provided a formal RAI required to complete its evaluation of the CR-3 EPU LAR (Reference 3).

Attachment A, "Response to Request for Additional Information – Nuclear Performance and Code Review Branch Technical Review of the CR-3 EPU LAR," provides the formal response to the RAI. Attachment A contains information that is considered proprietary. AREVA NP, Inc., as the owner of the proprietary information, has executed the affidavit provided in Attachment B and states that the identified information has been classified as proprietary, is customarily held in confidence, and not made available to the public. AREVA requests that the identified proprietary information be withheld from public disclosure in accordance with the provisions of

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

A001
NRR

10 CFR 2.390(a)(4). Attachment C, "Response to Request for Additional Information – Nuclear Performance and Code Review Branch Technical Review of the CR-3 EPU LAR (Non-proprietary)," is a non-proprietary copy of the formal RAI responses with the proprietary information redacted.

Attachment D, "List of Regulatory Commitments," includes a regulatory commitment to perform fuel design mechanical evaluations using the NRC approved COPERNIC fuel performance code and shift the TACO3 fuel burnup linear heat rate point to offset differences between the current and new fuel performance codes during finalization of the CR-3 EPU initial core reload analysis and prior to CR-3 EPU operation.

If you have any questions regarding this submittal, please contact Mr. Dan Westcott, Superintendent, Licensing and Regulatory Programs at (352) 563-4796.

Sincerely,

Jon A. Franke
Vice President
Crystal River Nuclear Plant

JAF/gwe

Attachments:

A. Response to Request for Additional Information – Nuclear Performance and Code Review Branch Technical Review of the CR-3 EPU LAR (Proprietary)

B. AREVA Affidavit for Withholding Proprietary Information from Public Disclosure

C. Response to Request for Additional Information – Nuclear Performance and Code Review Branch Technical Review of the CR-3 EPU LAR (Non-proprietary)

D. List of Regulatory Commitments

xc: NRR Project Manager
Regional Administrator, Region II
Senior Resident Inspector
State Contact

STATE OF FLORIDA

COUNTY OF CITRUS

Jon A. Franke states that he is the Vice President, Crystal River Nuclear Plant for Florida Power Corporation, doing business as Progress Energy Florida, Inc.; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Jon A. Franke
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 18TH day of JUNE, 2012, by Jon A. Franke.

Carolyn E Portmann

Signature of Notary Public
State of Florida

CAROLYN E. PORTMANN
Commission # DD 937553
Expires March 1, 2014
Bonded Thru Troy Fain Insurance 800-385-7019

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known ✓ -OR- Produced Identification ______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT B

AREVA AFFIDAVIT FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in Attachment A of the letter from Progress Energy Florida to the NRC entitled, "Crystal River Unit 3 - Response to Request for Additional Information to Support NRC Nuclear Performance and Code Review Branch (SNPB) Technical Review of the CR-3 Extended Power Uprate LAR (TAC No. ME6527)" dated June 18, 2012 and associated ANP-3120(P), Revision 0, entitled "Crystal River Unit 3 EPU – NRC RAI and Responses (LAR Sections 2.8.1, 2.8.2, 2.8.3)," dated June 2012 and referred to herein as "Documents." Information contained in these Documents has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these Documents be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in these Documents is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in these Documents has been made available,

on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 14th

day of June, 2012.

Kathleen A. Bennett

Kathleen Ann Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/15
Reg. # 110864

KATHLEEN ANN BENNETT
Notary Public
Commonwealth of Virginia
110864
My Commission Expires Aug 31, 2015

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT C

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION – NUCLEAR PERFORMANCE AND CODE REVIEW BRANCH TECHNICAL REVIEW OF THE CR-3 EPU LAR (NON-PROPRIETARY)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION – NUCLEAR PERFORMANCE AND CODE REVIEW BRANCH TECHNICAL REVIEW OF THE CR-3 EPU LAR (NON-PROPRIETARY)

By letter dated June 15, 2011, Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., requested a license amendment to increase the rated thermal power level of Crystal River Unit 3 (CR-3) from 2609 megawatts (MWt) to 3014 MWt (Reference 1). On March 21, 2012, via electronic mail, the NRC provided a draft request for additional information (RAI) to support the Nuclear Performance and Code Review Branch (SNPB) technical review of the CR-3 Extended Power Uprate (EPU) License Amendment Request (LAR). By teleconference on May 1, 2012 CR-3 discussed the draft RAI with the NRC to confirm an understanding of the information being requested. On May 4, 2012, the NRC provided a formal RAI required to complete its evaluation of the CR-3 EPU LAR.

This attachment is a non-proprietary copy of the formal RAI responses with the proprietary information redacted in accordance with the provisions of 10 CFR 2.390(a)(4). Redacted information is identified with braces { }.

For tracking purposes, each item related to this RAI is uniquely identified as SNPB X-Y, with X indicating the RAI set and Y indicating the sequential item number.

AREVA NP, Inc. (AREVA) Fuel Rod Performance Codes Background Information

Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Revision 7, (Reference 2) is approved for use by the NRC and provides descriptions of codes and methodology used to support of core reloads for Babcock and Wilcox (B&W) plants including CR-3. The approved TACO3/GDTACO fuel performance code family and the approved COPERNIC fuel rod performance code have been approved by the NRC for fuel rod analyses as documented in BAW-10179P-A and in Topical Report BAW-10186P-A, "Extended Burnup Evaluation," (Reference 3).

Historically, the TACO3/GDTACO code family has been used in; 1) fuel mechanical design evaluations, 2) loss of coolant accident (LOCA) initialization analyses, and 3) the input of thermal-hydraulic (T-H) evaluations utilizing the approved LYNXT code. The approved COPERNIC fuel performance code is currently used in B&W plant core reloads to support corrosion (i.e., oxide thickness) evaluations.

Beginning in 2009, augmentation factors for fuel temperature uncertainty were defined for the TACO3 and GDTACO codes to account for thermal conductivity degradation (TCD). The impact of reduced transient cladding strain (TCS) and centerline fuel melt (CFM) limits on peaking margins are evaluated as part of the cycle reload analyses with these augmentation factors applied to the linear heat rate (LHR) limits generated by TACO3 and GDTACO. At that time, parameter augmentation to account for TCD was applied in TACO3 and GDTACO when performing B&W core reload analyses, including CR-3 core reload analyses.

In late 2010, AREVA began implementing COPERNIC as the fuel rod performance code in analyses performed in the reload safety evaluations for B&W plants. For CR-3 EPU core reload

analyses, the COPERNIC code will replace the TACO3/GDTACO codes for fuel mechanical design evaluations and will continue to be used for corrosion evaluations. The following criteria analyses will be performed using the COPERNIC code:

a) TCS,
b) CFM,
c) rod internal pressure,
d) cladding fatigue,
e) cladding creep collapse, and
f) cladding oxidation.

For the CR-3 LOCA analyses, the TACO3/GDTACO code family uses methods to increase the fuel stored energy and, coupled with the burnup-dependent uncertainties applied to the predictions, partially compensate for TCD at moderate to high burnups. These methods, combined with the pin local to rod average EPU LOCA LHR burnup adjustments in the core reload analyses, continue to support the EPU LOCA initializations and T-H evaluations at all licensed fuel pin burnups.

The evaluation descriptions provided in Section 2.8.1, "Fuel System Design," of the CR-3 EPU Technical Report (TR) (Reference 1, Attachments 5 and 7) were based on the historic use of TACO3 and GDTACO. The supporting fuel analyses had been or were completed as the fuel TCD issue related to high fuel burnup was becoming an area of increased interest. The CR-3 EPU TR Section 2.8.1 text acknowledged the evolving issues related to fuel TCD; however, at this time, the CR-3 EPU core reload analyses have not been formally re-analyzed with the COPERNIC code. As indicated in Attachment D, "List of Regulatory Commitments," prior to CR-3 EPU operation, the CR-3 EPU initial core reload analysis will be performed using the COPERNIC code, which accounts for the effects of fuel TCD.

<u>**SNPB Requests for Additional Information**</u>

Section 2.8.1 Fuel System Design

1. (SNPB 1-1)

Section 2.8.1.2, *Description of Fuel Rod Analyses and Evaluations* states that "Augmentation factors were derived for TACO3 and GDTACO to correct cladding strain and centerline fuel melt limits for the effects of degradation of fuel thermal conductivity. Fuel rod internal gas pressure analyses were examined and were determined to not be adversely impacted."

(a) Provide detailed descriptions of evaluations, analyses and results for the derivation of augmentation factors for TACO3 and GDTACO to correct (1) cladding strain, (2) cladding fatigue, and (3) centerline fuel melt limits for the effects of degradation of fuel thermal conductivity with burn-up.

Response:

The statement in Section 2.8.1 of the CR-3 EPU TR (Reference 1, Attachments 5 and 7) regarding augmentation factors for TACO3 and GDTACO to account for the effects of degradation of fuel thermal conductivity degradation are no longer applicable for CR-3 fuel TCS, cladding fatigue, and CFM limits since fuel performance for EPU operating cycles will be analyzed using the NRC approved COPERNIC fuel performance code. The COPERNIC code explicitly models degradation of thermal conductivity resulting from fuel burnup, and does not require any correction to the analysis results in order to account for fuel TCD.

(b) Provide justification for the statement that the fuel rod internal gas pressure was determined not to be adversely impacted even after incorporating augmentation factors in TACO3 and GDTACO codes to correct for the thermal conductivity degradation (TCD) with burn-up.

Response:

The statement in Section 2.8.1 of the CR-3 EPU TR regarding the impact of TCD on fuel rod internal gas pressure analyses is no longer applicable since fuel performance for EPU operating cycles will be analyzed using the NRC approved COPERNIC fuel performance code. Fuel rod internal gas pressure analyses are performed using the COPERNIC code, which explicitly models degradation of thermal conductivity resulting from fuel burnup, and does not require any correction to the analysis results in order to account for fuel TCD.

2. (SNPB 1-2)

Appendix K to Title 10 of the *Code of Federal Regulations*, Part 50 – *ECCS [Emergency Core Cooling system] Evaluation Models*, Section I.A.1 stipulates that "The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.) To accomplish this, the thermal conductivity of the UO_2 [uranium dioxide] shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO_2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep."

(a) Explain how TACO3 predicted loss-of-coolant accident (LOCA) initialization fuel rod temperatures adequately account for the degraded thermal conductivity with burn-up for any peak cladding temperature limited analysis.

Response:

The BWNT LOCA evaluation model (EM) used for the CR-3 EPU LOCA analyses contains the NRC-approved steady-state fuel code TACO3 to supply initial fuel pin parameters and fuel stored energy for the UO_2 fuel LOCA peak cladding temperature (PCT) transient analyses. The TACO3 code includes consideration of the fuel pellet

density changes and expansion, cladding creep, fission gas release, and dynamic gap conductance changes with burnup. It also includes burnup-dependent gap gas composition variations that change the gap conductance. The GDTACO code, which is a TACO3 derivative code used for gadolinia doped fuel LOCA initializations, has similar models. Neither TACO3 nor GDTACO explicitly model the fuel thermal conductivity change with burnup.

The effect of the fuel TCD is accounted for in the TACO series of codes via hot spot uncertainty factors, burnup-dependent uncertainty factors, input initialization techniques that increase the fuel pin temperature inputs to the LOCA PCT analyses at all licensed local fuel pin burnups. The adequacy of the code method is demonstrated by benchmarking with time-in-life (TIL) specific data using a high power test rod.

The TACO3 code was developed in the 1980s. Steady-state fuel temperature benchmarks were originally performed with concentration on high LHR limits; however, the burnup ranges were limited. The statistical comparisons to the benchmark data showed that the code was nearly best-estimate and a fuel temperature uncertainty factor was developed to ensure a 95/95 fuel centerline temperature was predicted. This hot spot centerline fuel temperature uncertainty factor is { } percent and it is applied to the entire rod. Increases to the hot spot uncertainty factor at higher burnups were included to account for known TCD effects beginning in the early 1990s during the review of BAW-10186P-A (Reference 3).

The LOCA analyses use the rod average temperature from the steady-state fuel code (TACO3/GDTACO) as an input to reflect the initial fuel stored energy required by 10 CFR 50, Appendix K.I.A.1. This is partially accomplished by applying the hot spot centerline fuel temperature uncertainty factor established by benchmark comparisons to the volume-average fuel temperature. It is also applied to the entire length of the hot rod to increase the stored energy of the fuel pin. In addition, increased fuel temperature uncertainties are applied to TACO3 predictions to explicitly address TCD effects based on SIMFUEL data (References 4 and 5). Since 1994, an additional {

}. At a burnup of 62 GWd/mtU, the combined uncertainty using this approach is { } percent. The statistical uncertainty factor combined with the additive high burnup TCD-related uncertainty factor skew the predicted fuel temperatures to higher values.

The BWNT LOCA EM analyses that are performed span the entire range of licensed fuel pin burnup. For CR-3, the analyses are performed at three TILs: beginning of life (BOL); middle of life (MOL), between 34 to 45 GWd/mtU; and at end of life (EOL), 62 GWd/mtU. The EOL analyses are not PCT limited, but BOL and MOL analyses could be PCT-limited. The BOL analyses are not affected by TCD. The MOL and EOL analyses are affected by TCD and the method of analysis augments the uncertainty factors in accounting for it in the fuel stored energy term applied as input to the LOCA EM analyses.

TACO3 LOCA initializations use different inputs and methods for BOL versus TIL cases. The TACO3 code uses a UO_2 fuel thermal conductivity based on BOL fuel burnup for all

cases. BOL analyses use lower bound fuel enrichment input and minimum helium gas purity for the initial gas fill because these inputs maximize the fuel pin temperatures. The TIL analyses use a bounding high enrichment of approximately 5 weight percent along with a 7 to 10 percent increase to the bounding rod power history envelope. Both of these inputs increase the predicted fuel temperature with burnup and they result in more xenon gas release to the gap. Since xenon gas has a conductance that is approximately 20 times less than helium, it degrades the gap conductance and increases the fuel pellet temperatures, even after fuel-clad mechanical contact begins. The xenon degrades the net conductance under these conditions because the mechanical contact conductance does not dominate the TACO3 total gas conductance value. The TIL methods used for LOCA initialization also include imposing the Improved Technical Specifications (ITS) LOCA power peaking limit on the rod and holding it constant for 1000 MWd/mtU prior to the targeted rod-average burnup value. During this hold at the LOCA LHR limit, the fuel temperatures are increased, which releases additional xenon gas further increasing the fuel temperature by the end of the hold period. This predicted fuel temperature is augmented by the uncertainty factors described previously.

AREVA performed informal benchmark comparisons of TACO3 fuel temperature predictions to the developmental GALILEO code for the Mark-B-HTP fuel design. The GALILEO code explicitly models burnup-dependent fuel thermal conductivity. These comparisons further supported the conclusions that the TACO3 fuel temperature adjustments at higher rod burnups reasonably compensate for the TCD effects on fuel temperature inputs used in the LOCA PCT analyses.

A combination of benchmarking to prototypical data and code-to-code comparisons was used to assess the validity of TACO3 predictions with respect to TCD effects. TACO3 and GALILEO benchmarks were performed using test data from the Halden Boiling Water Reactor instrumented fuel assembly test IFA 432 Rod 3. This rod initially had a peak power of { } and its power decreased with burnup. It was selected for assessment because of its higher power level and moderate burnup ranges. The fuel pin temperatures at low burnups for the upper thermocouple { }.

The plotted data obtained for Rod 3 inlet thermocouple of the IFA 432 test reached a burnup of { }. This test fuel rod has a slightly larger cladding outer diameter than the Mark-B-HTP fuel pins but it had a reasonable initial gap size and fuel pellet densification. { }. The rod was instrumented with two centerline thermocouples located at the upper and lower regions of the rod. The rod power history, defined based on the irradiated power of the rod while in the reactor, was used as input to the GALILEO and TACO3 benchmark cases.

The lower thermocouple location of Rod 3 had lower powers than the top thermocouple location. Figure 1, "GALILEO Centerline Fuel Temperature Prediction of IFA 432-R3 (Lower Rod)," and Figure 2, "TACO3 Centerline Fuel Temperature Prediction of IFA 432-R3 (Lower Rod)," show the measured and predicted fuel centerline temperatures for the lower

thermocouple, for GALILEO and TACO3, respectively. The GALILEO benchmark had excellent comparison to the data with a measured to predicted temperature mean of { } with a standard deviation of { } for the lower thermocouple. The TACO3 benchmark statistics for the lower thermocouple was { } with a mean predicted minus measured difference of { }. As indicated in these figures, TACO3 fuel centerline temperature predictions were more conservative than GALILEO for the lower thermocouple.

The upper thermocouple location of Rod 3 had a higher power. Figure 3, "GALILEO Centerline Fuel Temperature Prediction of IFA 432-R3 (Upper Rod)," and Figure 4, "TACO3 Centerline Fuel Temperature Prediction of IFA 432-R3 (Upper Rod)," show the measured and predicted fuel centerline temperatures for the upper thermocouple, for GALILEO and TACO3, respectively. The GALILEO benchmark again had good agreement to the data with a measured to predicted temperature mean of { } and a standard deviation of { } for the upper thermocouple. The TACO3 benchmark statistics for the upper thermocouple was marginally better at { } with a mean predicted minus measured difference of { }. TACO3 fuel centerline temperature predictions again indicated more conservative than GALILEO for the upper thermocouple.

Figure 5, "TACO3 Measured/Predicted Temperatures versus BU for IFA 432-R3 (Lower Rod)," Figure 6, "TACO3 Measured/Predicted Temperatures versus BU for IFA 432-R3 (Upper Rod)," and Figure 7, "TACO3 Measured/Predicted Temperatures versus LHR for IFA 432-R3 (Upper Rod)," provide additional comparisons for the TACO3 predictions to the test data. These figures show the preponderance of measured-to-predicted ratio points are less than 1.0 indicating that TACO3 remains conservative with respect to fuel burnup and LHR. Given that TACO3 does not have a TCD model, it is expected that the prediction would lose margin or even become non-conservative as the pin burnup increases. However, the prediction remained conservative because of the xenon gas gap degradation predicted by TACO3 at the moderately high fuel temperatures experienced in the test rods. Figure 8, "TACO3 Predicted Temperatures versus LHR and BU for IFA 432-R3 (Upper Rod)," shows the TACO3 predicted centerline temperature with fuel burnup. This figure further delineates the predicted temperature points in distinct burnup ranges also showing an increase in fuel temperature predictions with burnup at a given LHR. As an example, at { }. At this low burnup range, TCD is not yet a key factor in the fuel pin temperature increase. { }. This fuel temperature increase predicted by TACO3 is created by the xenon gas release and degraded gap heat removal at high temperatures. So the xenon effect compensates for the TACO3 fuel temperature predictions even without a fuel pellet TCD model: thus, allowing TACO3 to replicate the fuel temperature data from test rods with actual TCD.

LOCA initialization cases at low LHR limits using TACO3 result in lower fuel temperatures with little to no xenon gas release. Therefore, the low power rods at high

burnups will be under predicted by TACO3 because of reduced xenon gas release and no fuel TCD model. However, LOCA analyses are run with rods at higher LHR limits where TACO3 has been shown to be more conservative relative to the fuel stored energy contributions. Therefore, use of TACO3 in the LOCA PCT analyses with the BWNT LOCA EM is considered acceptable because other methods and uncertainty factors compensate for the lack of an explicit TCD model for TIL cases.

As a result of recent nuclear industry interest related to TCD effects, further research and development efforts and investigations into options for replacing TACO3 identified that the TCD included in current state of the art codes is based on rod local burnup. However, the TACO3 uncertainty factors beyond { } were applied based on rod average burnups in the approved methodology. As indicated in Attachment D, "List of Regulatory Commitments," prior to CR-3 EPU operation, FPC will shift the TACO3 MOL fuel burnup LHR point to offset the local to rod-average burnup difference between the current and new AREVA fuel performance codes during finalization of the CR-3 EPU initial core reload analysis. This shift in LOCA burnup LHR limits was not included in the CR-3 EPU LOCA analyses; however, this shift in burnup preserves the fuel stored energy at MOL such that it does not change the LHR limits or the PCTs summarized in Section 2.8.5.6.3, "Emergency Core Cooling System and Loss-of-Coolant Accidents," of the CR-3 EPU TR (Reference 1, Attachments 5 and 7). The EOL LHR limits will be reduced by 0.5 kW/ft during the finalization of the CR-3 EPU initial reload analysis to account for the local to rod average burnup adjustments. The adjustments do not change the non-limiting PCT predictions for these cases.

The TACO3 code is used for UO_2 analyses and the GDTACO code is used to analyze gadolinia doped rods. Because of the similarity between these two codes and methods of analyses, the conclusions derived for the TACO3 code are directly applicable to the GDTACO code.

While the TACO3/GDTACO codes do not include TCD, the LOCA initialization method used for TIL cases mimics the fuel TCD effect on stored energy for LOCA applications. Use of conservative enrichments, a bounding power history curve, and holding the LOCA power peaking limit for 1000 MWd/mtU prior to the targeted rod average burnup value conservatively predicts higher fuel temperatures and fission gas release that degrades the gap gas conductance. Lower gap conductance elevates the fuel volume-average temperatures and this increase combined with the burnup-dependent fuel temperature uncertainty factor at burnups greater than { } offsets the TCD effects on the initial fuel temperatures used in LOCA analyses. Inputs from the TACO series of codes, developed with the techniques described herein along with burnup-dependent fuel temperature uncertainties applied to the adjusted TIL burnup ranges or LHR limits, are acceptable for use in the PCT-limited LOCA analyses for CR-3 operation up to the EPU power level.

Figure 1: GALILEO Centerline Fuel Temperature Prediction of IFA 432-R3 (Lower Rod)

Figure 2: TACO3 Centerline Fuel Temperature Prediction of IFA 432-R3 (Lower Rod)

Figure 3: GALILEO Centerline Fuel Temperature Prediction of IFA 432-R3 (Upper Rod)

Figure 4: TACO3 Centerline Fuel Temperature Prediction of IFA 432-R3 (Upper Rod)

Figure 5: TACO3 Measured/Predicted Temperatures versus BU for IFA 432-R3 (Lower Rod)

Figure 6: TACO3 Measured/Predicted Temperatures versus BU for IFA 432-R3 (Upper Rod)

Figure 7: TACO3 Measured/Predicted Temperatures versus LHR for IFA 432-R3 (Upper Rod)

Figure 8: TACO3 Predicted Temperatures versus LHR and BU for IFA 432-R3 (Upper Rod)

(b) If TACO3 or any other fuel performance codes used for the mechanical performance design, specifically TCD of the CR-3 fuel, lacks the capability to treat TCD, provide details of how the TCD is evaluated, when those were used.

Response:

Mechanical performance of the fuel design; i.e., fuel TCS, cladding fatigue, CFM limits, and internal gas pressure; will be evaluated using the NRC approved COPERNIC fuel performance code for the CR-3 EPU core reload analyses. The COPERNIC code explicitly models fuel TCD resulting from fuel burnup and does not require any correction to the analysis results in order to account for fuel TCD.

3. (SNPB 1-3)

Explain how the impact of TCD with burn-up on the non-LOCA transients and postulated accidents analyses have been implemented in, specifically but not limited to, the spectrum of control rod ejection accident analyses.

Response:

The non-LOCA analyses summarized in the CR-3 EPU TR (Reference 1, Attachments 5 and 7) were performed with the RELAP5/MOD-2-B&W system code. With the exception of the rod

ejection accident analysis, these non-LOCA analyses do not explicitly account for the impacts of TCD.

Accounting for Fuel TCD in Rod Ejection Accident Methodology

In an NRC to FPC letter dated January 28, 2010 (Reference 6), the NRC approved a plant-specific rod ejection accident methodology developed for CR-3 to analyze the rod ejection accident from EPU conditions (Reference 7). As stated in the NRC Safety Evaluation (SE) associated with the approval of the CR-3 rod ejection accident methodology, the COPERNIC code is used to provide fuel property input to both NEMO-K and LYNXT. The COPERNIC code explicitly models fuel TCD resulting from fuel burnup and does not require correction to the analysis results in order to account for fuel TCD. NEMO-K uses the fuel and clad thermal equations from COPERNIC and LYNXT uses input fitting tables versus temperature generated with COPERNIC. Both codes use a gap conductance fitting table generated with COPERNIC. Therefore, the rod ejection accident analysis explicitly models TCD with burnup for both the power response of the core (NEMO-K) and the thermal performance of the peak pin (LYNXT).

Accounting for Fuel TCD in Non-LOCA Methodology except Rod Ejection Accident

The RELAP5/MOD2-B&W non-LOCA methodology uses fuel thermal properties to calculate the fuel average temperatures and the heat flux for determining the feedback effects in the point kinetics model. Fuel temperatures predicted by steady-state NEMO calculations are used to initialize the non-LOCA models. The pellet-clad gap conductance is adjusted to achieve a core average fuel temperature consistent with the NEMO prediction. Topical Report BAW 10180-A, "NEMO – Nodal Expansion Method Optimized," (Reference 8) includes doppler coefficient benchmarks which indicate that the magnitude of the doppler coefficient is low. To obtain a better best-estimate model for the fuel temperature model in the steady-state NEMO model, the TACO3 fuel temperatures were biased. The TACO3 fuel temperatures with the bias are very similar to the COPERNIC values. No significant change is expected if TACO3 fuel temperatures with the bias are replaced by COPERNIC values.

The primary effect of TCD on non-LOCA analyses is the accelerated rate of fuel temperature increase during power increasing events which leads to a greater doppler feedback effect. Since the fuel and gap conductivities are balanced to arrive at the initial fuel temperature, repartitioning of the conductance between the fuel and the gap has a small effect on final results. As a result, it is concluded that the conservatisms inherent in the methodology utilized for the CR-3 EPU non-LOCA analyses summarized in the EPU TR are sufficient to address small differences induced by the fuel TCD, especially for slow power increasing transients.

During finalization of the CR-3 EPU initial reload analysis, fuel design mechanical performance inputs (i.e., fuel TCS and CFM limits) used in the CR-3 EPU non-LOCA analyses will be based on the NRC approved COPERNIC fuel performance code. The COPERNIC code explicitly models fuel TCD resulting from fuel burnup and does not require any correction to the analysis results in order to account for fuel TCD. COPERNIC-based limits have been implemented in the safety analysis for three operating B&W plants with no significant impact on Reactor Protection System (RPS) trip setpoints or plant operating limits.

4. (SNPB 1-4)

Provide details of the tests and the results from those tests that confirm that the spacer grids provide adequate support to maintain the fuel rods in coolable configuration under all conditions including EPU conditions for safe shutdown earthquake and LOCA.

Response:

Dynamic crush tests are performed to determine grid characteristics. The details of the spacer grid crush tests are provided in Section 3.3 of Addendum 1 of Topical BAW-10133P-A "Mark-C Fuel Assembly LOCA-Seismic Analyses," (Reference 9). As indicated in BAW-10179P-A, (Reference 2), the NRC staff has reviewed and approved the use of BAW-10133P-A in licensing applications for Mark-B fuel assembly designs. The grid external stiffness, damping and allowable impact loads are determined from these tests and used as input to the horizontal core seismic and LOCA analyses. Spacer grids are tested at operating temperatures.

The test spacer grids are filled with empty fuel rods and guide thimble segments. The test involves gradually increasing the impact velocity until grid instability takes place, which allows characterization of the grid behavior within the elastic range. The test results are proprietary and may be made available to NRC auditors.

Fuel assembly seismic loadings are not affected by EPU conditions. The maximum calculated grid impact load for the combined safe shutdown earthquake and LOCA loads considering EPU conditions are within the grid elastic load limit. With no permanent grid deformation predicted, the coolable geometry requirements are met for the fuel assemblies within the core. Thus, the spacer grid test results continue to assure that the spacer grids provide adequate support to maintain the fuel rods in a coolable configuration at EPU conditions.

5. (SNPB 1-5)

Provide details of the fuel rod bow evaluations performed to assure that the Mark-B-high thermal performance (HTP) fuel design is bounded by the rod bow criteria established under the plant design conditions of the CR-3 EPU.

Response:

The fuel rod bow is determined by the examination of irradiated assemblies based on the fuel assembly design and plant operating conditions. Fuel rod bow alters the design pitch dimensions between adjacent rods. Fuel rod bow is determined based on a statistical analysis of water channel gaps per Topical Report BAW-10147P-A, "Fuel Rod Bowing in Babcock & Wilcox Fuel Designs," (Reference 10). A departure from nucleate boiling ratio (DNBR) penalty and power peaking uncertainty are applied to conservatively account for fuel rod bow as a function of burnup level. The DNBR penalty and power peaking uncertainty are not specifically dependent on plant operation at EPU conditions.

FPC, AREVA, and other B&W-design plant owners monitor fuel rod bow or water channel gap closure to verify fuel rod bowing remains below the acceptance criteria established in BAW-10147P-A.

The fuel rod bow is influenced by cell type (i.e., fuel rod in adjacent cell, guide tube in adjacent cell, or peripheral cell), grid slip load, grid relaxation, fuel rod creep rate, and cross flow forces. For EPU conditions, parameters affecting rod bowing remain the same except cross flow forces. The cross flow forces used in the generic Mark-B-HTP fuel rod bow analyses conservatively bound the calculated cross flow forces in the CR-3 core at EPU conditions.

Section 2.8.2 Nuclear Design

6. (SNPB 1-6)

Describe the analysis procedure used to ensure that the shutdown margin for Modes 1 and 2 operations is within the Technical Specification limits throughout the transition and equilibrium cycle of EPU operation at CR-3. Specifically, show how the eigenvalue biases and uncertainties are determined and accounted for during the transition cycles.

Response:

The analysis process used to ensure that the shutdown margin (SDM) is within the CR-3 ITS limits during Modes 1 and 2 operation is described in Section 5.4.2.1, "Shutdown Margin," of BAW-10179P-A (Reference 2), and the core physics model used to perform the SDM evaluations is described in BAW-10180A (Reference 8). Regarding the SDM analysis performed for each CR-3 reload cycle, Section 5.4.2.1 of BAW-10179P-A states, in part:

This SDM analysis process, including applicable uncertainties in available and required rod worth, requires no additional penalties or uncertainties for EPU conditions. Further, the reactivity parameters which comprise the SDM calculations for Modes 1 and 2 are independent of the eigenvalue biases. Instead, the SDM is based upon core reactivity differences between pre- and post-reactor trip conditions, not the eigenvalue itself.

Eigenvalue biases are determined through analysis of plant reactivity performance based on the most recent operating cycles. In this reactivity performance analysis, measured plant startup and operational data is compared to the predicted data from the NEMO core simulator. The latest measured data obtained through startup physics testing and cycle operation is incorporated into the eigenvalue biases prior to each reload evaluation. The resulting eigenvalue biases are incorporated into the NEMO physics model of the given reload. Startup and operational data for CR-3 EPU cycles will be included in the eigenvalue bias update process for each subsequent CR-3 EPU cycle. In addition, the correspondence of the uncertainties used in reload licensing analysis and the startup physics test acceptance criteria provide an important linkage to ensure the CR-3 EPU cycles are operated within the licensing basis.

Section 5.5, "Startup Physics Testing," of BAW-10179P-A describes how the startup physics tests account for the accuracy and uncertainties of the NEMO physics model:

Based on AREVA reload experience, the NEMO code has accurately modeled a wide range of core designs and power levels, including reactor cores:

- with higher power densities than the CR-3 EPU conditions;
- which have gone through power uprates; and
- which use gadolinia as a burnable absorber.

During transition cycles when no operational data is yet available for operation at EPU conditions, startup physics testing provides verification of the accuracy of the NEMO physics model. The following provides a description of the interrelationship of the startup test and reload licensing analysis as stated in Section 5.5 of BAW-10179P-A:

Table 5-2, "Reload Startup Physics Testing for B&W-Designed Plants," of BAW-10179P-A provides a list of plant startup of physics tests. The relevant tests and acceptance criteria from

Table 5-2 of BAW-10179P-A which are used to demonstrate the accuracy of the eigenvalue biases and SDM parameters are:

These startup physics tests are currently performed at CR-3 and will continue to be performed at CR-3 to ensure that the SDM for Modes 1 and 2 operations is within the ITS limits throughout the transition and equilibrium cycles of EPU operation at CR-3.

7. (SNPB 1-7)

Provide the calculations that support the statement that "Shutdown margin requirements can be satisfied with increases in boric acid volumes in the Boric Acid Storage Tank(s)."

Response:

SDM in Modes 3, 4, and 5 specified in CR-3 ITS 3.1.1, "SHUTDOWN MARGIN (SDM)," is ensured by borating the Reactor Coolant System (RCS) to a specific shutdown boron concentration. Calculations for the CR-3 EPU cycles indicate that the shutdown boron concentration required to maintain SDM in Modes 3, 4, and 5 for the EPU cycles has increased substantially relative to pre-EPU cycles. Based on checks of the boric acid storage tank (BAST) requirements, it was determined that the existing volume and concentration requirements would be inadequate for the EPU cycles. Through iteration, new BAST requirements were established to ensure adequate borated water is available for boration to achieve the required SDM in Modes 3, 4, and 5. Cycle-to-cycle variability in the volume requirements was considered when establishing new bounding volume and concentration requirements for the BAST(s). The ability to meet the shutdown requirements using the minimum required boric acid volumes and concentrations of the BAST(s) is confirmed each cycle as part of the reload licensing analysis.

The following equations were used to evaluate the required BAST volumes and times to achieve the required SDM:

(1)

(2)

The following calculations are provided to demonstrate the BAST capabilities in a scenario at the beginning of cycle prior to the EPU conditions. This example is based on the example provided in the current CR-3 ITS Bases B 3.1.1.

The following is a similar beginning of cycle calculation performed considering EPU conditions to show how a typical EPU cycle will meet the revised BAST requirements of achieving the required SDM. The method of determining the system mass was refined in the EPU calculation to more accurately reflect system density differences within components versus using a common density for all components. As a result, the system mass value is lower in the following EPU example calculation than the pre-EPU example calculation.

{ }

Based on the results of this calculation, the revised BAST requirements for CR-3 EPU conditions continue to ensure the required SDM requirements can be achieved for Modes 3, 4, and 5.

8. (SNPB 1-8)

Provide the details of a typical calculation in which margins to power peaking limits based on centerline fuel melt, transient cladding strain, and steady-state departure from nucleate boiling criteria were calculated to evaluate the reactor protection system (RPS) axial offset limits that would be input to the determination of the RPS power/imbalance/flow trip function.

Response:

Section 5.3.1, "Axial Power Imbalance Protective Limits," of NRC-approved topical report BAW-10179P-A (Reference 2) describes the methodology used to calculate margins to power peaking limits based upon CFM, TCS, and steady-state departure from nucleate boiling (SS-DNB) criteria and to determine the axial offset protective limits that are the basis for determination of the RPS Nuclear Overpower RCS Flow and Measured Axial Power Imbalance trip setpoints. This RPS trip function provides fuel protection with respect to peaking factors that could result in violation of the CFM, TCS, or DNB fuel design limits during normal operation and anticipated operational occurrences.

Peaking margins are determined in a cycle-specific three-dimensional core power distribution (maneuvering) analysis. In the maneuvering analysis, limiting core power distributions are simulated with an approved core nodal design code and {

}. The NRC-approved NEMO nuclear design code described in BAW-10180-A (Reference 8) is used for CR-3 core design and reload safety evaluation, including the power distribution calculations for the EPU core maneuvering evaluations described in Section 2.8.2, "Nuclear Design" of the CR-3 EPU TR (Reference 1, Attachments 5 and 7).

Design analyses employ nodal core design codes that typically simulate axial power distributions in units of dimensionless peaking factors and core axial offset. However, the CR-3 instrumentation and control systems are designed to measure axial power imbalance. Consequently, parameters and limits defined in offset units are converted to imbalance units for

use in documents such as the CR-3 ITS and the Core Operating Limits Report (COLR). Both terms are measures of the difference in integrated power in the top half and bottom half of the core. Offset and imbalance are related by the following equation:

%Offset = %Imbalance / fraction of rated thermal power (RTP)

The evaluation performed for the CR-3 EPU core designs utilized the ranges shown in Table 1, "Parameter Values used in Maneuvering Analysis for Conceptual EPU Cycles," for the parameters in the NEMO power distribution simulations.

Table 1: Parameter Values used in Maneuvering Analysis for Conceptual EPU Cycles

{ }, these are typical ranges of parameters that are used in standard cycle reload analyses for B&W plants. CR-3 currently operates with the APSR group in the fully withdrawn position during power operation. {

}.

{

}. Since the purpose of the evaluation is to demonstrate operational capability of the EPU conceptual cycles within applicable safety criteria, cases at design overpower conditions of 112% RTP were evaluated. Since margins to the peaking limits are larger at reduced thermal powers, it is not necessary to simulate power distributions at the complete set of thermal power levels that would be executed in a reload safety evaluation for licensing purposes.

Following completion of the core power distribution simulations, margins of each power distribution to the peaking limits are computed. Peaking margin is defined as:

{ }

{

}. SS-DNB limits are typically specified as maximum allowable peaking (MAP) limits. CFM and TCS limits are usually specified as LHR limits in kW/ft units. {

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Table 2, "Axial Offsets that Define RPS Offset Protective Limits at 112% RTP – EPU Conceptual Cycle 18", provides a summary of the axial offset limits for the conceptual CR-3 EPU Cycle 18 as defined by the margin versus offset correlations determined in the maneuvering analysis evaluation.

As shown in Table 2, the limiting axial offset values at 112% RTP for CR-3 EPU Conceptual Cycle 18 are; { }. The axial offset limits shown in bold type are the basis of the values for CR-3 EPU Conceptual Cycle 18 specified in Table 2.8.2-3, "EPU versus Cycle 16 Offset and LCO Rod Insertion Limits," of the CR-3 EPU TR (Reference 1, Attachments 5 and 7), and are applicable to 4-pump operation at design overpower conditions.

Table 2: Axial Offsets that Define RPS Offset Protective Limits at 112% RTP – EPU Conceptual Cycle 18

	{ }	{ }	{ }
Negative Offset Limit, %	**- 35.9**	{ }	(See Note 1)
Positive Offset Limit, %	{ } (See Note 2)	{ } (See Note 3)	**+ 33.5**
Time in Cycle, EFPD	{ }	{ }	{ }

Note 1: { }

Note 2: { }

Note 3: { }

The limiting EPU axial offset values determined by this process form the basis for calculation of the RPS axial power imbalance protective limits that will be specified in the CR-3 COLR for the licensed EPU fuel cycle. This process is repeated each cycle for the CR-3 core reload safety evaluation. { }. Application of the error adjustments, which are applied to the RPS offset protective limits to derive the allowable and actual trip setpoint envelopes, is described in more detail in Section 7.3, "Power/Imbalance/Flow Trip," of BAW-10179P-A (Reference 2).

9. (SNPB 1-9)

Discuss the evaluations which indicated that the magnitudes of RPS offset limits, limiting condition for operation offset limits and shutdown margin-based rod insertion limits in some instances were more restrictive than previous CR-3 fuel cycles. Explain why these limits are still acceptable for normal operation and anticipated operational transients for the EPU operation at CR-3.

Response:

RPS and Limiting Conditions for Operation (LCO) axial offset limits were calculated at design overpower conditions for the EPU core reload analyses using the NRC approved methods described in BAW-10179P-A (Reference 2).

Both axial offset and axial power imbalance are measures of the difference in power in the top half and bottom half of the core. Axial offset values are not adjusted for instrumentation errors. Offset is converted to imbalance using the equation below and the resulting imbalance is then adjusted for instrumentation errors.

% offset = % imbalance / fraction of RTP
Thus, % offset = % imbalance at 100% RTP.

To determine the impact of EPU operation on the CR-3 RPS and LCO axial offset limits, calculations were performed at design overpower conditions, and the resulting axial offset limits were compared to the actual Cycle 16 (i.e., pre-EPU) axial offset limits at overpower conditions. From this comparison, the EPU negative axial offset limits were found to be more restrictive as indicated in Table 2.8.2-3 of the CR-3 EPU TR (Reference 1, Attachments 5 and 7). The actual Cycle 16 axial offset limits have margin to the actual axial imbalance RPS trip setpoint envelope and the LCO axial imbalance operating limits specified in the Cycle 16 COLR. Table 3, "Comparison of EPU offset and Imbalance Limits to Cycle 16 Limits," provides the comparison of the EPU negative axial offset limits and estimated axial imbalance limits to the Cycle 16 COLR limits.

Table 3: Comparison of EPU offset and Imbalance Limits to Cycle 16 Limits

RPS						
	Cycle 16	%RTP at 2609 MWt		Limiting EPU Cycle	%RTP at 3014 MWt	Delta
Negative Offset Limit (%)	{ }	110.2		{ }	112.0	{ }
Allowable Negative Imbalance Trip Setpoint (%)	-33.7	108.0		-23 (est.)	108.0	-11
Actual Negative Imbalance Trip Setpoint (%)	-16.5	106.5		---	---	---

LCO						
	Cycle 16	%RTP at 2609 MWt		Limiting EPU Cycle	%RTP at 3014 MWt	Delta
Negative Offset Limit (%)	{ }	100.4		{ }	100.4	{ }
Allowable Negative Imbalance Operating Limit (%)	-23.8	100.4		-12 (est.)	100.4	-12
COLR Negative Imbalance Operating Limit(%)	-17.0	100.4		---	---	---

From Table 3 it can be seen that the EPU negative axial offset RPS limit is not expected to have an impact on the actual RPS Axial Power Imbalance Trip Setpoint Envelope; however, the EPU negative axial offset LCO limit is expected to have an impact on the COLR Axial Power Imbalance Operating Limit Envelope. The calculated EPU negative axial offset LCO limit is impacted by the fuel TCD adjustments to the peaking limits; adjusted TACO3 LHR limits were used in the determination of the LCO axial offset limits. {

}. Although the RPS offset limits also used adjusted TACO3 LHR limits, the impact is not as significant. As required by ITS 5.6.2.18, "CORE OPERATING LIMITS REPORT (COLR)," during finalization of the CR 3 EPU initial reload analysis, the current COLR Axial Power Imbalance Operating Limit

Envelope will be evaluated and adjusted as necessary; and, although not expected to change, the actual RPS Axial Power Imbalance Trip Setpoint Envelope will also be evaluated and adjusted if necessary to account for instrumentation errors (e.g., instrument drift).

As indicated in Section 2.8.2 of the CR-3 EPU TR (Reference 1, Attachments 5 and 7), successful operation of CR-3 at the EPU power level with the more restrictive negative axial offset LCO limit will be enhanced by installing a new Makeup System tank bypass line. This new bypass line will improve the reactivity control response time for RCS soluble boron concentration changes. The faster reactivity control response time as a result of this plant modification will improve the ability to make reactivity adjustments to control axial offset when the plant operates with the Integrated Control System (ICS) in automatic mode.

Therefore, it is concluded that the RPS and LCO axial offset limits at EPU conditions are acceptable for normal operation and anticipated operational transients for EPU operation at CR-3.

SDM rod insertion limits (RILs) were calculated for EPU conditions using the NRC approved calculation methods described in BAW-10179P-A (Reference 2). These limits were compared to the SDM RILs for Cycle 16 and found to be more restrictive. As indicated in Table 2.8.2-4, "End of Cycle SDM-Based Rod Insertion Limits (RIL)," of the CR-3 EPU TR (Reference 1, Attachments 5 and 7), the Cycle 16 error adjusted SDM RIL is 195% rod index and the most limiting EPU error adjusted SDM RIL is 270% rod index, where rod index is the sum of the withdrawal of Regulating Rod Groups 5, 6, and 7. Thus, a rod index of 270% corresponds to Group 7 at 70% withdrawn. The operating RILs specified in the COLR for normal operation ensure the power peaking factors remain valid, while the SDM-based RILs specified in the COLR ensure sufficient inserted reactivity is available in the control rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth rod is fully withdrawn upon trip.

As a result of the change in the SDM RILs calculated for the EPU conditions, the operating RILs for the regulating rod banks will be adjusted to provide rod insertion margin to the SDM RILs. To evaluate the impact of adjusting the operating RILs, the expected response of the regulating rod groups during EPU operation was evaluated. Specifically, a main turbine runback from EPU power level was simulated to determine if the more restrictive RILs could be accommodated. A main turbine runback was considered since it represents one of the most challenging operational transients to the RILs. This simulation demonstrated that the SDM RILs will not be exceeded during the limiting anticipated operational transient condition. As previously described, the Makeup System tank bypass line will also improve the ability to control regulating rod group position when the plant operates with the ICS in automatic mode. Therefore, it is concluded that the SDM RILs at EPU conditions are acceptable for normal operation and anticipated operational transients for EPU operation at CR-3.

10. (SNPB 1-10)

For the control rod ejection accident analysis, provide details of how the impact of fuel TCD with burn-up has been implemented. Explain the impact of TCD on core stored energy (energy deposition in fuel).

Response:

The fuel performance of the limiting power peaking fuel pin during an ejected rod accident is evaluated with LYNXT using { } fuel properties calculated by COPERNIC. The COPERNIC code explicitly models fuel TCD resulting from fuel burnup and does not require correction to the analysis results in order to account for fuel TCD. The details of how COPERNIC results have been implemented for a rod ejection accident at EPU conditions are provided in Topical Report ANP-2788P, "Crystal River 3 Rod Ejection Accident Methodology Report," (Reference 7, Attachment E). The NRC staff concluded that the CR-3 specific rod ejection accident methodology is acceptable in an NRC to FPC letter dated January 28, 2010 (Reference 6). The impact of TCD with burnup is included in the resultant energy depositions, temperatures, and DNBRs of the rod ejection accident analysis summarized in Section 2.8.5.4.6, "Spectrum of Rod Ejection Accidents," of the CR-3 EPU TR (Reference 1, Attachments 5 and 7).

11. (SNPB 1-11)

An examination of NEMO-K topical report, BAW-10221P indicates that its fuel pin temperature distribution calculation is based on a one-dimensional heat conduction equation with a fuel thermal conductivity as a function of just temperature and not fuel burn-up also. Discuss the impact on the results from NEMO-K due to the non-dependence of thermal conductivity with fuel burn-up.

Response:

The model for the fuel pin temperature distribution in Topical Report BAW-10221P-A, "NEMO-K A Kinetics Solution in NEMO," (Reference 11) uses thermal material properties that are a function of burnup. As stated on page 3-12 of BAW-10221P-A:

> *"The thermal conductivity correlations are functions of the mesh temperature, material content, nodal burnup, and initial plutonium content."*

Section 6.1 of ANP-2788P (Reference 7, Attachment E) clarifies which thermal conductivity correlations are used in NEMO-K. As stated in the NRC SE associated with the approval of the CR-3 rod ejection accident methodology described in ANP-2788P (Reference 6), the COPERNIC code is used to provide fuel property input to NEMO-K. Also, NEMO-K uses a gap conductance fitting table generated with COPERNIC. Therefore, the fuel pin temperature distribution calculation in NEMO-K for the CR-3 rod ejection accident explicitly includes the impact of fuel TCD with burnup and does not require a correction to the analysis results to address TCD.

Section 2.8.3 Thermal and Hydraulic Design

12. (SNPB 1-12)

Provide details of the steady-state analyses performed to assess the steady-state performance of the EPU core at CR-3, for expected nominal and overpower conditions and for a varying number of reactor coolant pumps in operation.

Response:

The analyses assessing the steady-state performance of the CR-3 EPU core were performed in accordance with the NRC approved methodologies and codes defined in Section 6 of BAW-10179P-A (Reference 2) within the limitations and restrictions described in the associated NRC SEs. These methodologies and codes are the same as those used to support current CR-3 operating cycles.

CR-3 Steady-State Analyses Overview

- { } in Topical Report BAW-10187P-A, "Statistical Core Design for B&W Designed 177 FA Plants," (Reference 12);
- { };
- { } ITS 2.1.1, "Reactor Core SLs;"
- { }; and
- { }

CR-3 Steady-State Analyses Details

The following provides details of the analyses assessing steady-state performance of the CR-3 EPU core.

As stated in Section 6.4, "Reactor Coolant System DNB Safety Limits," of BAW-10179P-A:

{ }

The CR-3 RCS DNB safety limits are based on the SCD methodology in BAW-10187P-A (Reference 12). The limits and associated uncertainties specified in BAW-10187P-A are shown as an example of the methodology and the exact values used for the CR-3 EPU are shown in Table 2.8.3-1, "Thermal-Hydraulic Design Parameters for CR-3," and Table 2.8.3-2, "CR-3 SCD Input Variables," of the CR-3 EPU TR (Reference 1, Attachments 5 and 7). .

{ }

{ }.

{ }.

As stated in Section 6.6.1 "Calculation of Safety Limit MAPs (RPS MAPS)," of BAW-10179P-A (Reference 2):

{

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{

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Figure 9: CR-3 EPU Pressure-Temperature Limits

As indicated in the CR-3 EPU LAR (Reference 1, Attachment 2), a new Safety Limit Figure (ITS Figure 2.1.1-1) matches the P-T limits calculated by the DNB analyses at EPU conditions and ensures that the P-T operating region will preclude the reactor from exceeding a safety limit when operating up to the EPU power level. Because the 4-pump safety limit curve is the most limiting, it was selected as the ITS Safety Limit figure and used to determine the RPS RCS variable low pressure trip values.

13. (SNPB 1-13)

(a) Provide details of the process in which the statistical design limit (SDL) for departure from nucleate boiling ratio (DNBR) is established.

Response:

The SDL for the DNBR was established in accordance with the NRC approved methodologies and codes defined in Section 6 of BAW-10179P-A, (Reference 2) within the

limitations and restrictions described in the associated NRC SEs. The same SCD methodology and codes have been used to support operating cycles at CR-3 since 1994. The SDL is a DNBR limit reflecting the statistical treatment of uncertainties which allows the use of nominal values to be used during a DNBR analyses for those parameters whose uncertainties are covered under the SDL. The Mark-B-HTP fuel assembly utilizes two CHF correlations based on spacer grid design in two different axial regions of the fuel assembly; below the lowermost Mark-B HTP intermediate grid, the BWC CHF correlation is used; and from the leading edge of the lowermost Mark-B HTP intermediate grid to the top of active fuel, the BHTP CHF correlation is used. Although the process for determining an SDL is applicable to both axial regions, the details and example values that are shown are associated with the primary axial region where the BHTP CHF correlation is applied.

The following description from Section 6.2.9 "Statistical Core Design," of BAW-10179P-A is an overview of the process used to establish the SDL:

> *"The design philosophy for core departure from nucleate boiling protection originally followed a deterministic approach where uncertainties that affect the minimum DNBR are simultaneously assumed to be at their worst-case values. The minimum core DNBR is calculated using compounding of the uncertainties, for comparison with the DNBR design limit associated with the applicable CHF correlation.*
>
> *"A more realistic assessment of core DNB protection, called Statistical Core Design (SCD), has been developed by application of statistical techniques to treat the core state and bundle uncertainties. SCD is a widely accepted method that is utilized to reduce some of the undue conservatism of traditional methods, while still allowing for the traditional compounding of variables not amenable to statistical treatment.*
>
> *"A response surface model was used to obtain an overall uncertainty on the calculated DNBR. The response surface model was based on a full central composite design method in order to reduce uncertainty in the response surface model fit. The uncertainty distribution for each of the applicable variables was subjected to a Monte Carlo propagation analysis to determine an overall statistical DNBR uncertainty, which was used to establish a Statistical Design Limit (SDL). The SDL is higher than the CHF limit upon which it is based, because it contains allowances for all of the propagated uncertainties as well as the uncertainty on the original CHF correlation. When the minimum DNBR is calculated, the variables treated statistically are entered into the LYNXT thermal-hydraulic calculations at their nominal levels. Variables not treated in deriving the SDL continue to be entered at their most adverse allowable levels.*
>
> *"Generic uncertainty allowances included in the SCD methodology for the B&W 177-FA plants are described in [BAW-10187P-A]."*

The DNBR uncertainty values are plant specific and Table 2.8.3-2 of the CR-3 EPU TR (Reference 1, Attachments 5 and 7) provides the calculated DNBR uncertainty values. The

SDL meets the following DNB protection criteria: 1) a hot pin SDL DNB protection of 95%/95% and 2) a core wide SDL DNB protection of 99.9%.

The response surface model (RSM) developed for CR-3 is sufficiently conservative to bound the parameters and uncertainties provided in Table 2.8.3-1 and Table 2.8.3-2 of the CR-3 EPU TR (Reference 1, Attachments 5 and 7). The RSM, { } was used in the Monte Carlo propagation. The SDL values, for hot pin DNBR protection and core wide DNBR protection, are determined at various core { } because the DNBR sensitivity for the uncertainties may be different for various operation conditions. The maximum SDL value observed for hot pin DNBR and core-wide DNBR protection of the CR-3 core considering various core states was determined to be { }. For each CR-3 core reload, the statistical DNBR uncertainties are re-verified for the existing SDL or, if appropriate, new uncertainty allowances are applied requiring the re-calculation of a new SDL value.

(b) Provide details of the process in which the thermal design limit (TDL) for DNBR is achieved.

Response:

The TDL for the DNBR is established using the process described in Section 6 of BAW-10179P-A (Reference 2). The following description from Appendix F, "Exit Limited SCD Analysis," of BAW-10187P-A is an overview of the process to establish the TDL.

{ }

Figure 10, "Graphical Presentation of Pre- and EPU DNBR Limits," provides a graphical presentation of the DNBR limits and margin of the current CR-3 cycle (i.e., pre-EPU) and the EPU cycle. The magnitude of the TDL retained margin is established on the need for such margin in future operation. The fuel design is not changing as a result of the CR-3 EPU; therefore, the margin of the DNBR-limiting anticipated operational transient, loss of forced reactor coolant flow, to the TDL value as shown between the two DNBR value scales in Figure 10 is reduced { }.

Figure 10: Graphical Presentation of Pre- and EPU DNBR Limits

(c) Explain how the margin between the TDL and the SDL is used to offset cycle-to-cycle abnormalities (such as transition core effects or deviations in uncertainty). Also, explain how the margin between the TDL and SDL is utilized to provide flexibility in the fuel cycle design.

Response:

As described in Section 6.2.9 of BAW-10179P-A (Reference 2), the difference between the TDL and SDL is known as the retained margin. Specifically, Section 6.2.9 of BAW-10179P-A states:

> *"The retained margin is available to offset penalties, such as transition core effects, or deviations in uncertainty values from those incorporated in the SDL, or to provide flexibility in the fuel cycle design."*

Section 6.3.3 "Hydraulic Compatibility" of BAW-10179P-A provides additional details regarding the retained margin:

{

}

The retained margin is available to offset penalties, such as transition core effects, or deviations in uncertainty values from those incorporated in the SDL, or to provide flexibility in the fuel cycle design. Examples of the DNB penalties that may be incurred and offset portions of the retained margin are:

- DNB transition core,
- debris in the core,
- process errors,
- spacer grid damage,
- DNBR margin needs for a transient event, and
- other cycle-specific penalties.

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General

14. (SNPB 1-14)

The NRC staff intends to run FRAPCON-3.4 benchmark calculations of the Mark-B-HTP 15x15 fuel rod design (NUREG/CR-7022, Volume 1, "FRAPCON-3.4: A Computer Code for the Calculation of Steady-State Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burn-up," and Volume 2, "FRAPCON-3.4: Integral Assessment," Office of Nuclear Regulatory Research, US NRC, March 2011). Please provide the following input.

A. Rod Power History, kilowatt per foot (KW/ft), as a function of gigawatt-days per metric ton of uranium (GWd/MTU)

   1. Bounding thermal-mechanical operating envelope (e.g., radial falloff curve)
   2. Discuss any application of rod power uncertainties
   3. Include power histories for different pellet designs (UO_2, Gadolinium).

B. Axial Power Distribution (AXPD) (F_z at each axial node)

   1. Include AXPDs for different axial blanket configurations.

C. Fuel Rod Design Specifications and Manufacturing Tolerances

   1. Outer diameter
   2. Inside diameter
   3. Pellet diameter
   4. Stack length
   5. Plenum length
   6. Pellet height
   7. Dish radius
   8. Dish depth
   9. Spring outside diameter
   10. Spring wire diameter
   11. Number of spring turns
   12. Maximum uranium-235 (U-235) enrichment (%)
   13. Average U-235 enrichment (%)
   14. Maximum gadolinia content (%)
   15. Water in pellet, parts per million (ppm)
   16. Nitrogen in pellet (ppm)
   17. Pellet density (% theoretical density)
   18. Open porosity (%)
   19. Pellet surface roughness (microns)
   20. Expected density increase (grams/cubic centimeter, gms/cc)
   21. Sintering temperature (°F)

22. Cladding Alloy = (Material name)
23. Final thermal treatment of the cladding alloy
24. Cladding surface roughness (microns)
25. Cladding texture factor
26. Cladding Hydrogen content (ppm)
27. Fill gas pressure
28. Fill gas composition
29. Rate of CRUD accumulation factor (mils/hr)
30. CRUD thermal conductivity

D. Coolant conditions

1. Coolant inlet temperature (°F)
2. Coolant mass flux (lbm/hr-ft^2)
3. System pressure (pounds per square inch absolute, psia)

Response:

As stated in Section 1.0, "Introduction to the Crystal River Unit 3 Extended Power Uprate Technical Report," of the CR-3 EPU TR (Reference 1, Attachments 5 and 7), the Mark-B-HTP fuel design is the same design currently in service at CR-3. The Mark-B-HTP fuel design was first introduced at CR-3 in Cycle 14.

As indicated in Attachment D, "List of Regulatory Commitments," prior to CR-3 EPU operation, the CR-3 EPU initial core reload analysis will be performed using the COPERNIC code for fuel design mechanical evaluations. The COPERNIC code is a modern fuel performance code which explicitly models fuel thermal conductivity degradation resulting from fuel burnup. The calibration database for the COPERNIC code consists of a wide variety of fuel designs and operating conditions. The database spans the entire approved burnup range and extends to high power levels that include powers from fuel melt experiments. The power levels considered in the COPERNIC validation database bound the EPU power levels at CR-3. The corrosion database includes "hot" reactors with high coolant temperatures and high power fuel management. Based on the range of data included in the validation database, the COPERNIC fuel performance code is applicable for use in fuel mechanical design evaluations of the fuel design and core operating conditions representative of the CR-3 EPU core.

As documented in the NRC SE for the COPERNIC fuel performance code in NRC to Framatome ANP letter dated June 14, 2002, "Framatome ANP Topical Report BAW-10231P, 'COPERNIC Fuel Rod Design Computer Code' – Correction of Error in Safety Evaluation (TAC No. MA6792)," (Reference 13), audit calculations were performed with the FRAPCON-3 fuel performance code for the Mark-B fuel rod design as part of the NRC staff review of the COPERNIC code topical report. The Mark-B-HTP fuel rod design currently utilized in the CR-3 core is similar to the Mark-B fuel rod design used for the audit calculations of COPERNIC. Additionally, the design parameters of the Mark-B-HTP fuel rod are within the range of parameters considered in the COPERNIC validation database. The audit calculations included comparisons to COPERNIC calculations for maximum rod internal pressure, LOCA temperatures (stored energy) and pressures, temperatures for the fuel melting, and clad strain

analysis. Based on these comparisons, the NRC staff concluded that the COPERNIC code is acceptable for application to the various analyses. As such, FPC believes that FRAPCON-3.4 confirmatory calculations of the Mark-B-HTP 15x15 fuel rod design are unnecessary and the requested fuel design information is not provided.

References

1. FPC to NRC letter dated June 15, 2011, "Crystal River Unit 3 – License Amendment Request #309, Revision 0, Extended Power Uprate." (ADAMS Accession No. ML112070659)

2. AREVA Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Revision 7, January 2008. (Proprietary)

3. AREVA Topical Report BAW-10186P-A, "Extended Burnup Evaluation," Revision 2, June 2003. (Proprietary)

4. P.G. Lucuta, et. al., "Thermal Conductivity of SIMFUEL," J. Nucl. Mater. 188, 1992, pp. 198-204.

5. P.G. Lucuta, et. al., "Task 1.4 Thermal Properties of U02 – Based SIMFUEL," Belgonucleaire High Burnup Chemistry Report, HBC 92/43, Nov. 1992.

6. NRC to FPC letter dated January 28, 2010, "Crystal River Unit 3 Nuclear Generating Plant – Issuance of Amendment Regarding Adoption of a New Methodology for Rod Ejection Accident Analysis Under Extended Power Uprate Conditions (TAC NO. ME0730)." (ADAMS Accession No. ML093280945)

7. FPC to NRC letter dated February 26, 2009, "Crystal River Unit 3 – License Amendment Request #307, Revision 0 Methodology for Rod Ejection Accident Analysis Under Extended Power Uprate Conditions." (ADAMS Accession No. ML090700533)

8. AREVA Topical Report BAW-10180-A, NEMO – Nodal Expansion Method Optimized," Revision 1, March 1993.

9. Addendum 1 to Topical Report BAW-10133-A, "Mark-C Fuel Assembly LOCA-Seismic Analyses," Revision 1, October, 2000. (ADAMS Accession No. ML003767624)

10. AREVA Topical Report BAW-10147P-A, "Fuel Rod Bowing in Babcock & Wilcox Fuel Designs," Revision 1, May 1983. (Proprietary)

11. AREVA Topical Report BAW-10221P-A, "NEMO-K A Kinetics Solution in NEMO," Revision 0, September 1998. (Proprietary)

12. AREVA Topical Report BAW-10187P-A, "Statistical Core Design for B&W Designed 177 FA Plants," Revision 0, March 1994. (Proprietary)

13. NRC to Framatome ANP letter dated June 14, 2002, "Framatome ANP Topical Report BAW-10231P, 'COPERNIC Fuel Rod Design Computer Code' – Correction of Error in Safety Evaluation (TAC No. MA6792)." (ADAMS Accession No. ML021360461)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT D

LIST OF REGULATORY COMMITMENTS

List of Regulatory Commitments

The following table identifies those actions committed to by Florida Power Corporation (FPC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please notify the Superintendent, Licensing and Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

Regulatory Commitment	Due date/event
During finalization of the Crystal River Unit 3 (CR-3) Extended Power Uprate (EPU) initial core reload analysis, FPC will perform fuel design mechanical evaluations using the NRC approved COPERNIC fuel performance code and shift the TACO3 middle-of-life fuel burnup linear heat rate point to offset the local to rod-average burnup difference between the current and new AREVA fuel performance codes.	Prior to CR-3 EPU operation.