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W3F1-2007-0037

August 2, 2007

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

SUBJECT: License Amendment Request NPF-38-271
to Support Next Generation Fuel
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests an amendment to Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specification (TS) 6.9.1.11.1, Core Operating Limits, TS 3.5.1, Safety Injection Tanks, and TS 3.6.1.5, Containment Air Temperature. The proposed changes will add new analytical methods and will modify the containment average air temperature and safety injection tank level to support the implementation of Next Generation Fuel.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards consideration. The bases for these determinations are included in the attached submittal.

The proposed change includes new commitments as summarized in Attachment 3. A similar change to add the new analytical methods for Core Operating Limits is requested for Arkansas Nuclear One, Unit 2 (ANO-2).

Entergy requests approval of the proposed amendment by March 14, 2008 in order to support the spring 2008 refueling outage. Once approved, the amendment shall be implemented prior to startup following the spring 2008 refueling outage. Although this request is neither exigent nor emergency, your prompt review is requested.

A001

NRR

If you have any questions or require additional information, please contact Ron Williams at 504-739-6255.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 2, 2007.

Sincerely,

KTW/DM

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. List of Regulatory Commitments

cc: Dr. Bruce S. Mallett
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Attachment 1

W3F1-2007-0037

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford 3).

The proposed change will revise the following Waterford 3 Technical Specifications (TS):

- TS 6.9.1.11.1, Core Operating Limits Report (COLR), by adding new analytical methods that will be used to determine the core operating limits;
- TS 3.5.1, Safety Injection Tanks, by lowering the maximum Safety Injection Tank (SIT) level; and
- TS 3.6.1.5, Containment Air Temperature, by raising the minimum allowable containment air temperature.

2.0 PROPOSED CHANGE

The proposed change will modify TS 6.9.1.11.1 by adding the following new analytical methods:

15) "CE 16 x 16 Next Generation Fuel Core Reference Report," WCAP-16500-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-16500-P, Revision 0, "CE [Combustion Engineering] 16x16 Next Generation Fuel [(NGF)] Core Reference Report," (Methodology for Specification 3.1.1.3 for Moderator Temperature Coefficient (MTC), 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, 3.2.4.b, 3.2.4.c, and 3.2.4.d for DNBR Margin, and 3.2.7 for ASI).

16) "Optimized ZIRLO™," WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A, (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

17) "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," WCAP-16523-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR), WCAP-16523-P, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," (Methodology for Specification 3.2.4.b, 3.2.4.c, and 3.2.4.d for DNBR Margin).

18) "ABB Critical Heat Flux Correlations for PWR Fuel," CENPD-387-P-A (Methodology for Specification 3.2.4.b, 3.2.4.c, and 3.2.4.d for DNBR Margin and 3.2.7 for ASI).

19) "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood" CENPD-132, Supplement 4-P-A, Addendum 1-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) CENPD-132 Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE [Combustion Engineering] Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than

1 in/sec Core Reflood" (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

The use of these analytical methods is associated with the implementation of Combustion Engineering (CE) 16 x 16 Next Generation Fuel (NGF) as defined in WCAP-16500-P.

Administrative changes are proposed on TS pages 6-20a and 6-20b in that text that is currently at the bottom of page 6-20a associated with TS 6.9.1.11.2 and 6.9.1.11.3 will be moved to page 6-20b as will some of the above listed references. No content change is proposed. This change is administrative in nature; no further discussion is included.

Next generation fuel by itself impacts the large break loss of coolant accident (LBLOCA) emergency core cooling system (ECCS) performance peak cladding temperature (PCT) results. The ECCS calculations are impacted by the increase in the core hydraulic pressure loss, the increase in the core cross-sectional flow area, and the decrease in the fuel rod cladding outside diameter. In particular, the core reflood calculations during a LBLOCA are impacted by the NGF design changes due to lower reflood rates and resulting lower reflood heat transfer coefficients for the hot rod. The following TS changes are proposed to accommodate this impact.

- Maximum allowable SIT level will be lowered from 77.8% to 72.8% (TS 3.5.1)
- Minimum allowable containment average air temperature will be raised from 90°F to 95°F (TS 3.6.1.5). The action statement will also be modified to address actions associated with containment average air temperature greater than or equal to 90°F and less than 95°F.

In summary, changes are proposed to:

- add new analytical methods to the list of COLR references (TS 6.9.1.11.1);
- lower the maximum allowable SIT level (TS 3.5.1); and
- raise the minimum allowable containment average air temperature and modify the action statement (TS 3.6.1.5).

These changes are needed to support implementation of the CE 16 x 16 NGF design.

3.0 BACKGROUND

Combustion Engineering 16 x 16 NGF as defined in WCAP-16500-P will be implemented at Waterford 3 beginning in Cycle 16 commencing after the spring 2008 refueling outage. The fuel design is intended to provide improved fuel reliability by reducing grid-to-rod fretting issues, improved fuel performance for high duty operation, and enhanced operating margin.

3.1 <u>Core Operating Limits Report</u>

The major analysis methods for Waterford 3 remain unchanged as a result of the implementation of CE 16 x 16 NGF. Those areas that are impacted are identified in several topical reports (TRs), which are described below.

WCAP-16500-P and Final Safety Evaluation

Westinghouse TR WCAP-16500-P describes the methods and models that will be used to evaluate the acceptability of CE 16 x 16 NGF at CE plants.

The WCAP provides a comparison between CE 16 x 16 standard and the CE 16 x 16 NGF fuel assemblies and documents that the two assembly types are mechanically compatible. In addition, Nuclear Design, Thermal and Hydraulic Design, Safety and Setpoint, and Structural evaluations were performed demonstrating the CE 16 x 16 NGF fuel assembly design is acceptable for use at Waterford 3.

The TR requested approval of the CE 16 x 16 NGF design to 62 MWd/kgU peak rod average burnup for use in CE NSSS units using the current CE Reload methodology. No change is proposed to the current Waterford 3 licensing bases which restricts peak rod average burnup to 60 MWd/kgU.

WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A

The CE 16 x 16 NGF assembly design contains Optimized ZIRLO™ clad fuel rods. Entergy has submitted by letter (Reference 1) a proposed change to Waterford 3 TS 5.3.1 Fuel Assemblies, to add Optimized ZIRLO™ as a fuel rod cladding material. An exemption request to apply the acceptance criteria of 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," to Optimized ZIRLO™ was also submitted (Reference 1).

WCAP-16523-P and Final Safety Evaluation

This TR describes the departure from nucleate boiling (DNB) correlations that will be used to account for the impact of the CE 16 x 16 NGF fuel assembly design. The correlation was developed to accurately reflect thermal performance of the NGF design with the side-supported vane grids and multiple grid spacing.

CENPD-387-P-A

This TR provides the departure from nucleate boiling correlation that will be used to evaluate the DNB impact of non-mixing vane grid spans for CE 16 x 16 standard and NGF assemblies.

CENPD-132, Supplement 4-P-A, Addendum 1-P and Final Safety Evaluation

The addendum provides an optional steam cooling model that can be used in the ECCS Performance Evaluation to support the implementation of the CE 16 x 16 NGF fuel assembly design. The optional steam cooling model is being used to support the implementation of CE 16 x 16 NGF assemblies in Waterford 3. The results of this analysis will be submitted in a separate letter.

3.2 Safety Injection Tank Water Level

The SITs are used to flood the core with borated water following depressurization as a result of a Loss of Coolant Accident (LOCA) and/or Main Steam Line Break (MSLB). A lower SIT water level results in a larger gas volume which raises the driving force for water into the

Reactor Coolant System (RCS) as RCS pressure lowers. This higher SIT flow rate lowers the PCT for a LBLOCA.

3.3 Containment Average Air Temperature

The limitation on containment minimum average air temperature ensures that the ECCS is capable of maintaining a PCT less than or equal to 2200°F during LBLOCA conditions. A higher containment average air temperature results in a higher post accident containment pressure, a higher reflood rate, and therefore a lower PCT.

4.0 TECHNICAL ANALYSIS

4.1 Core Operating Limits Report

WCAP-16500-P and Final Safety Evaluation (SE)

"CE 16 x 16 Next Generation Fuel Core Reference Report," WCAP-16500-P has been approved by the NRC (Reference 7). Entergy meets the Limitations and Conditions contained in the NRC SE as follows.

1. *Using approved methods, the licensee must ensure that all of the design criteria specified in TR WCAP-16500-P are satisfied on a cycle-specific basis (SE Section 3.3.1).*

   As part of the reload methodology, all of the new design criteria specified for CE 16 x 16 NGF per WCAP-16500-P, Table 1-1 will become part of the reload analysis basis. Using approved models and methods, the reload analysis, which is reviewed per the requirements of 10 CFR 50.59, will check/confirm that these design criteria are met.

2. *Fuel assembly component design and configuration (e.g., type and distribution of spacer grids and IFM grids) are limited to the five designs described in TR WCAP-16500-P and in response to RAI No. 2 (SE Section 3.2).*

   The Waterford 3 NGF assembly is consistent with the Plant B design defined in Figure 1-1 of WCAP-16500-P and for the Plant B design documented in the response to RAI No. 2 of Reference 2.

3. *The reference fuel assembly design, CE 16 x 16 NGF, its fuel mechanical design methodology and design criteria, are approved up to a peak rod average burnup of 62 GWd/MTU. A fuel burnup limit may exist, either explicitly or implicitly, in other portions of a plant's licensing basis. The NRC staff's approval of this topical report allows the CE 16 x 16 NGF assembly to reach a rod average burnup of 62 GWd/MTU. However, a license amendment request, specifically addressing each plant's licensing basis including radiological consequences, is required prior to extending burnup beyond current levels. Further, the NRC staff's SE for Optimized ZIRLO™ (Addendum 1 to TR WCAP-12610-P-A and TR CENPD-404-P-A) specified a 60 MWd/kgU burnup limit and this limitation must be revised prior to extending the peak rod average burnup for the NGF design (SE Section 3.4).*

The current Waterford 3 licensing basis restricts peak rod average burnup to 60 MWd/kgU. Entergy is not proposing a change to this limit.

4. *Licensees shall demonstrate the accuracy of their growth predictions based upon measured data and this validation shall be ahead of the burnups achieved by batch implementation. The growth model validation (e.g., measured versus predicted) should be documented in a letter(s) to the NRC (SE Section 3.2.1).*

   The growth data presented in Figure 2-15 of WCAP-16500-P Supplement 1-P is ahead of the projected exposure for the first cycle implementation of NGF. The fluence for the data is approximately 7 x 10^{21} nvt, which corresponds to an assembly average burnup of 39 MWD/kgU. The projected end of first cycle assembly average burnup is approximately 27 MWD/kgU. As indicated in the responses to RAIs 1a and 1b in Reference 2, additional growth data will be obtained from future Lead Test Assembly (LTA) exams ahead of the exposure achieved by batch implementation. This data will be provided to the NRC as it becomes available.

5. *To compensate for NRC staff concerns related to the digital setpoints process, an interim margin penalty of 6 percent must be applied to the final addressable constants (e.g., BERR1* 1.06, [(1+EPOL2)*1.06 - 1.0]) calculated following the 1/64 hypercube setpoints process (Response No. 6 of Reference 6). Removal of this interim margin penalty will be considered after the digital setpoints methods have been formalized, documented (e.g., revision to TR WCAP-16500-P), and approved by the NRC (SE Section 3.7).*

   For the first cycle of Waterford 3 that contains a batch of NGF (Cycle 16), the analysis that calculates the uncertainty addressable constants for the Core Operating Limit Supervisory System (COLSS) on-line monitoring system and the Core Protection Calculator (CPC) System will not account for the NGF design and Critical Heat Flux (CHF) correlations. Therefore, the resultant DNB uncertainty addressable constants will not credit the DNB margin gain due to NGF, will not require application of the interim 6% margin penalty and will not require use of the 1/64 hypercube setpoints process.

   Full DNB margin credit for NGF will begin with the next cycle (Cycle 17) where the NGF CETOP-D model with the WSSV-T and ABB-NV CHF correlations will be used in the COLSS and CPC uncertainty analyses. The Modified Statistical Combination of Uncertainties (MSCU) analysis performed each cycle, as described in Reference 3, will automatically calculate appropriate DNB uncertainty addressable constants for COLSS and CPC reflecting the DNB margin impact of NGF. The 1/64 hypercube setpoints process as well as other process steps described in response to RAI 6 of WCAP-16500-P Supplement 1-P (Reference 4) will be utilized in this analysis. In addition, the 6% interim margin penalty will be applied to the resultant addressable constants until its removal has been approved by the NRC.

6. *Licensees are required to demonstrate that during transition cores, DNB margin gains associated with the NGF design offset (1) any impacts of flow starvation due to increased pressure drop and (2) uncertainty associated with predicting local flow characteristics. Further, licensees must detail the analytical methods and results of their transition core LOCA and non-LOCA analyses (SE Sections 3.7 and 3.10).*

First time engineering implementation analyses have been performed for transition and full cores of NGF. The analytical methods are defined in WCAP-16500-P for NGF implementation. For the transition cycle the COLSS on-line monitoring system and the CPC system will continue to utilize the current models and the CE-1 CHF correlation. For the first cycle of Waterford 3 that contains a batch of NGF (Cycle 16), the potential DNB margin gain, after accounting for the flow redistribution, is expected to be 12%. This margin gain is sufficient to compensate for any negative impact of the mixed core of NGF and standard fuel on transient and setpoint analyses. For non-LOCA analysis of the transition core NRC approved analytical methods are applied and results are based on a CETOP-D model for standard fuel so there is no transition core impact on transients. Full DNB margin credit for NGF will begin with the next cycle (Cycle 17) where the WSSV-T and ABB-NV DNB correlations will be used in the NGF CETOP-D model. The transition core LOCA evaluations for ECCS Performance including the implementation of CE 16 x 16 NGF assemblies are being finalized and will be submitted to NRC for review.

7. *Implementation of CE 16 x 16 NGF assemblies necessitate re-analysis of the plant-specific LOCA analyses. Licensees are required to submit a license amendment containing the revised LOCA analyses for NRC review. Upon approval, the revised LOCA analyses constitute the analysis-of-record and baseline for which future changes will be measured against in accordance with 10 CFR 50.46(a)(3) (SE Section 3.7).*

   The revised LOCA analyses for ECCS Performance including the implementation of CE 16 x 16 NGF assemblies for full core configuration are being finalized and will be submitted to NRC for review. It should be noted that the introduction of NGF does not impact the post-LOCA long term cooling analysis. Upon approval and implementation of NGF, these revised LOCA analyses will constitute the new analysis-of-record and baseline for Waterford 3.

8. *Using approved models and methods, Westinghouse will continue to limit peak local power experienced during Condition I and II events to ensure that fuel temperature remains below melting temperature at all burnups. This evaluation may be both plant and cycle-specific (SE Section 3.3.4).*

   Peak local power experienced during Condition I and II events will be limited to ensure fuel temperature remains below melting temperature at all burnups in accordance with Waterford 3 Technical Specification 2.1.1.2. This will be confirmed during each reload analysis.

9. *The NRC staff's approval of TR WCAP-16500-P establishes the licensing basis for batch implementation of the CE 16 x 16 NGF assembly design. Licensees wishing to implement this fuel design are required to submit a license amendment request, where applicable, updating their Core Operating Limits Report list of methodologies with the "A" version of this TR.*

   This amendment request satisfies this condition, however, the "A" version of the TR is not yet published.

10. *The NRC staff's review did not include the LOCA model changes described in Appendix A of TR WCAP-16500-P. Therefore, a licensee will have to submit a license amendment, if they desire to use The Appendix A LOCA model changes.*

Changes to the LOCA model outlined in Appendix A of TR WCAP-16500-P were resubmitted to the NRC by Westinghouse under CENPD-132, Supplement 4-P-A, Addendum 1-P, and have been approved for use in license amendment applications as described below.

<u>WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A</u>

Entergy submitted by letter (Reference 1) a proposed change to Waterford 3 TS 5.3.1 Fuel Assemblies, to add Optimized ZIRLO™ as a fuel rod cladding material. Entergy's compliance with the Limitations and Conditions of this TR are included in Reference 1.

<u>WCAP-16523-P-A and Final Safety Evaluation</u>

"Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," WCAP-16523-P has recently been approved by the NRC (Reference 5). Entergy meets the Limitations and Conditions contained in the final NRC SE as follows.

1. *The WSSV correlation must be used in conjunction with the VIPRE code since the correlation was developed based on VIPRE and the associated VIPRE input specifications. Other uses of the WSSV correlation should reference this TR and be based on appropriate benchmarking with VIPRE.*

   This condition is not applicable to Waterford 3 as the WSSV correlation with VIPRE will not be used at this time.

2. *The WSSV-T correlation must be used in conjunction with the TORC code since the correlation constants were developed based on TORC and the associated TORC input specifications. The correlations may also be used in the CETOP-D code in support of reload design calculations benchmarked by TORC.*

   The WSSV-T correlation is used in conjunction with TORC and CETOP-D codes in support of reload design calculations.

3. *The WSSV and WSSV-T correlations must also be used with the optimized Tong Fc shape factor for non-mixing and side-supported mixing vane grids to correct for non-uniform axial power shapes.*

   The optimized Tong Fc shape factor was utilized for non-mixing and side-supported mixing vane grids in the Waterford 3 NGF Thermal Hydraulic (TH) implementation analyses.

4. *The range of applicability for both the WSSV and the WSSV-T correlations are:*

Parameter	*Units*	*Range*
Pressure	*psia*	*1,495 to 2,450*
Local coolant quality	--	*≤ 0.34*
Local mass velocity	*10^6 lbm/hr-ft^2*	*0.90 to 3.46*
Matrix heated hydraulic diameter, Dhm	*inches*	*0.4635 to 0.5334*
Heated hydraulic diameter ratio, Dhm/Dh	--	*0.679 to 1.00*
Heated length, HL	*inches*	*48* to 150*
Grid spacing	*inches*	*10.28 to 18.86*
** Set as minimum HL value, applied at all elevations below 48 inches.*		

The WSSV-T correlation was applied according to Section 6.2 of WCAP-16523-P within the above range of applicability in the Waterford 3 NGF TH implementation analyses.

CENPD-387-P-A

The following conditions are satisfied when applying the ABB-NV correlation for non-mixing vane grid spans for CE 16 x 16 Standard and NGF assemblies:

1. *The ABB-NV and ABB-TV correlations indicate a minimum DNBR limit of 1.13 will provide a 95 percent probability with 95 percent confidence of not experiencing CHF on a rod showing the limiting value.*

   The ABB-NV correlation is applied for non-mixing vane grid spans for CE 16 x 16 standard and NGF assemblies. The minimum DNBR correlation limit of 1.13 is used. The WSSV-T correlation is applied for the mixing vane grid spans of the NGF fuel as described in Section 6.2 of WCAP-16523-P instead of the ABB-TV correlation.

2. *The ABB-NV and ABB-TV correlations must be used in conjunction with the TORC code since the correlations were developed on the basis of the TORC and the associated TORC input specifications. The correlations may also be used in the CETOP-D code in support of reload design calculations.*

   The ABB-NV correlation for non-mixing vane grid spans for CE 16 x 16 standard and NGF assemblies is used in conjunction with both TORC and CETOP-D codes.

3. *The ABB-NV and ABB-TV correlations must also be used with the ABB-CE optimized F_c shape factor to correct for non-uniform axial power shapes.*

   The ABB-NV correlation will be used with the ABB-CE optimized F_c shape factor to correct for non-uniform axial power shapes.

4. *Range of applicability for the ABB-NV and ABB-TV correlations:*

Parameter	*ABB-NV Range*	*ABB-TV Range*
Pressure (psia)	*1750 to 2415*	*1500 to 2415*
Local mass velocity (Mlbm/hr-ft^2)	*0.8 to 3.16*	*0.9 to 3.40*
Local quality	*-0.14 to 0.22*	*-0.10 to 0.225*
Heated length, inlet to CHF location (in)	*48 to 150*	*48 to 136.7*
Grid Spacing (in)	*8 to 18.86*	*8 to 18.86*
Heated hydraulic diameter ratio, Dhm/Dh	*0.679 to 1.08*	*0.679 to 1.000*

The specified range of applicability for the designated parameters is used when applying the ABB-NV correlation.

5. *The ABB-NV and ABB-TV correlation will be implemented in the reload analysis in the exact manner described in Section 7.1 of Topical Report CENPD-387-P, Revision 00-P.*

   The ABB-NV correlation is applied according to Section 7.1 of CENPD-387-P-A for non-mixing vane grid spans for CE 16 x 16 Standard and NGF assemblies. The WSSV-T correlation is applied for the mixing vane grid spans of the NGF fuel as described in Section 6.2 of WCAP-16523-P instead of the ABB-TV correlation.

6. *Technology transfer will be accomplished only through the process described in Reference 5 which includes ABB-CE performing an independent benchmarking calculation for comparison to the licensee generated results to verify that the new CHF correlations are properly applied for the first application by the licensee.*

   There is no technology transfer between Westinghouse and Entergy at this time.

CENPD-132, Supplement 4-P-A, Addendum 1-P and Final Safety Evaluation

"Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model, Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood," CENPD-132, Supplement 4-P-A, Addendum 1-P has recently been approved by the NRC (Reference 6).

The conclusion of the SE stated that Limitations and Conditions 3, 4, and 5 are appropriate for use when evaluating CE 16 x 16 NGF design fuel assemblies. The first two Limitations and Conditions included in the SE for CENPD-132, Supplement 4-P-A, Addendum 1-P are for fuel designs other than CE 16 x 16 NGF. Entergy meets the Limitations and Conditions contained in the final NRC SE as follows.

3. *Limitation on the Optional Steam Cooling Heat Transfer Model*

   The result of the grid model enhancement cannot result in the use of a heat transfer coefficient greater than FLECHT. The FLECHT upper-bound heat transfer coefficient, *as required by the current NRC licensing constraint, is also applied to the spacer grid optional steam cooling model improvement.*

The computer code logic for the optional steam cooling heat transfer model in the STRIKIN-II hot rod heatup computer code contains a specific algorithm to insure that the current NRC licensing constraint on the use of the FLECHT upper-bound heat transfer coefficient is also applied to the spacer grid steam cooling model improvement calculated in the PARCH steam cooling module. Therefore, this Limitation and Condition is automatically satisfied when performing the Waterford 3 licensing calculations using the version of the STRIKIN-II computer code containing the approved optional steam cooling heat transfer model.

4. *Use of the Optional Steam Cooling Model*

If a licensee wants to use the optional steam cooling model, then a license amendment request should be submitted including the analyses performed to determine its applicability to the specific fuel design being evaluated, as discussed in Section 3.3.1, 3.3.2, and 3.3.3 above. In addition, the licensee should provide the results of the evaluation with and without the optional steam cooling model, in a format similar to the graphical results provided in the reference calculations presented in the supplemental TR. The peak cladding temperature, local oxidation, and steam cooling flow rates should be included in the submittal. These comparisons will enable the NRC staff to confirm the acceptability of the use of the optional steam cooling model.

Since the optional steam cooling model is being used for Waterford 3 ECCS Performance Analyses, then a license amendment request will be submitted including the analyses and comparison graphical results needed to confirm the acceptability of the use of the optional steam cooling model.

5. *Use of Flow Blockage and Reynolds Number Limits (Section 3.3.3)*

For use of this topical report at a specific plant, the flow blockage and Reynolds number limits, as discussed in Section 3.3.3 above, should be confirmed by plant-specific analyses.

The computer code logic for the optional steam cooling heat transfer model in the PARCH module of the STRIKIN-II hot rod heatup computer code contains specific computational constraints to print warning and diagnostic output messages to alert the user if the calculation is found to be outside the range of applicability for flow blockage and Reynolds number. Therefore, this Limitation and Condition is automatically satisfied when performing the Waterford 3 licensing calculations using the version of the STRIKIN-II computer code containing the approved optional steam cooling heat transfer model.

4.2 Safety Injection Tank Water Level

The allowed range of SIT volume is being reduced by lowering the maximum level from 77.8% to 72.8%. This change reduces the calculated PCT for a LBLOCA due to its impact on SIT flow rate and the resulting core reflood rate. LBLOCA analyses are performed using the maximum SIT level allowed to ensure conservative results. The lower maximum SIT level results in a larger initial gas volume in the SIT. As the SIT injects during the LBLOCA, the gas pressure in the SIT is reduced less which provides a slightly greater driving force for SIT water to be injected. This higher water flow rate refills the reactor vessel more rapidly. The higher core reflood rate results in less time for the cladding to heat up and better core reflood heat transfer coefficients and therefore lower PCTs. This change is a more restrictive initial condition and is done to offset the impact to the calculated LBLOCA results of the NGF fuel design by itself.

Although the small break LOCA (SBLOCA) analyses use the SIT level as an input, the limiting results are not affected by actual level used. The limiting SBLOCA results occur for the largest break size where the SITs do not inject prior to the time of PCT.

No other accident analysis uses SIT maximum level as an input.

4.3 Containment Average Air Temperature

The allowable range of containment air temperature is being reduced by raising the minimum temperature limit from 90°F to 95°F. This change reduces the calculated PCT for a LBLOCA due to its impact on containment pressure and the resulting reflood rate. LBLOCA analyses are performed using the lowest allowed containment temperature to ensure conservative results. The higher minimum containment temperature results in higher initial containment heat sink temperatures and higher initial SIT temperatures. This reduces the amount of energy that the heat sinks and SITs can remove from the steam in the containment, which results in a slightly higher containment pressure during the LOCA. The higher containment pressure results in better reflood rates and therefore lower PCT.

If the containment temperature falls below the proposed minimum TS limit and remains above 90°F, then an appropriate reduction in peak linear heat generation rate, to be defined in the COLR, will be required.

No other accident analyses use minimum containment temperature as an input. Thus, LBLOCA is the only accident analysis that is affected by this change.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

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

Title 10 of the *Code of Federal Regulations* (10 CFR) Paragraph 50.36(c)(2)(ii) requires that Technical Specifications (TS) limiting conditions for operation be established for process variables, design features, and operating restrictions for which a value is assumed as an initial condition of a design basis accident in the licensee's safety analyses. As such, amendments may be required for each fuel cycle to update the values of cycle-specific parameter limits in the TSs. To eliminate the need for an amendment to update the cycle-specific parameter limits for each fuel cycle while complying with 10 CFR 50.36(c)(2)(ii) requirements, the U.S. Nuclear Regulatory Commission (NRC) has allowed licensees to use an alternative incorporating the cycle-specific parameter limits in the COLR. Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," provides the COLR implementation guidance, which includes the requirement to list the NRC-approved analytical methods used to confirm the safety of the core operating limits in the TSs. The analytical methods referenced in the TSs would identify the topical reports (TRs) by number, title, and date, or identify the staff's safety evaluation report for a plant-specific methodology by NRC letter and date. To further avoid the need for TS changes every time a revision to an approved TR is approved, the staff approved TS Task Force (TSTF) Traveler TSTF-363, "Revised Topical Report Reference in ITS 5.6.5, COLR," which allows for the listing of the TRs in the TS by only the numbers and titles, with the detailed identification of the TR revisions, supplement numbers, and approval dates specified in the COLR. Implementation of this TSTF has been approved for Waterford Steam Electric Station, Unit 3 (Waterford 3).

The proposed change adds several Westinghouse TRs which have been reviewed and approved by the NRC for licensing application. The addition of these TRs to Waterford 3 TS follows the same guidance of GL 88-16 and TSTF-363 in that only the TR numbers and titles will be listed in the TS, with the detailed identification of the report revisions and approval dates specified in the COLR.

By letter (Reference 1) Entergy has requested an exemption to 10 CFR 50.46 and Appendix K in support of implementing Optimized ZIRLO™. No other exemptions or relief from regulatory requirements, other than the TS, are required to support the proposed changes.

5.2 <u>No Significant Hazards Consideration</u>

The proposed change will modify the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specification (TS) related to the analytical methods that will be used to confirm the safety of core operating limits by adding the following references to topical reports:

- "CE 16 x 16 Next Generation Fuel Core Reference Report," WCAP-16500-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-16500-P, Revision 0, "CE [Combustion Engineering] 16x16 Next Generation Fuel [(NGF)] Core Reference Report"
- "Optimized ZIRLO™," WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A,
- "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," WCAP-16523-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR), WCAP-16523-P, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes"
- "ABB Critical Heat Flux Correlations for PWR Fuel," CENPD-387-P-A
- "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less than

1 in/sec Core Reflood" CENPD-132, Supplement 4-P-A, Addendum 1-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) CENPD-132 Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE [Combustion Engineering] Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood"

Use of the referenced methodologies will support implementation of Combustion Engineering (CE) 16 x 16 Next Generation Fuel (NGF). The fuel design is intended to provide improved fuel reliability by reducing grid-to-rod fretting issues, improved fuel performance for high duty operation, and enhanced operating margin.

Changes are also proposed to the maximum allowable safety injection tank level and the minimum allowable containment average air temperature. These values are used as inputs to the large break loss of coolant accident (LBLOCA) and small break loss of coolant accident (SBLOCA) analyses. The revised allowable values for both parameters result in a reduction to the calculated peak cladding temperature (PCT) for a LBLOCA and have no impact to the small break loss of coolant accident (SBLOCA) analysis results.

Entergy Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   Core Operating Limits Report (COLR)

   The proposed changes to the COLR TS are administrative in nature and have no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident. Changes to the calculated core operating limits may only be made using NRC approved methodologies, must be consistent with all applicable safety analysis limits, and are controlled by the 10 CFR 50.59 process.

   The proposed change will add the following topical reports to the list of referenced core operating analytical methods.

   <u>WCAP-16500-P and Final Safety Evaluation (SE)</u>

   Westinghouse topical report WCAP-16500-P describes the methods and models that will be used to evaluate the acceptability of CE 16 x 16 NGF at CE plants. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met. Prior to implementation of NGF the new core design will be analyzed with applicable NRC staff approved codes and methods.

WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A

The proposed change allows the use of methods required for the implementation of Optimized ZIRLO™ clad fuel rods. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

WCAP-16523-P and Final Safety Evaluation

This topical report describes the departure from nucleate boiling correlations that will be used to account for the impact of the CE 16 x 16 NGF fuel assembly design. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met. Prior to implementation of NGF the new core design will be analyzed with applicable NRC staff approved codes and methods.

CENPD-387-P-A

The proposed addition of this topical report provides the departure from nucleate boiling (DNB) correlation that will be used to evaluate the DNB impact of non-mixing vane grid spans for CE 16 x 16 standard and NGF assemblies. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

CENPD-132, Supplement 4-P-A, Addendum 1-P and Final Safety Evaluation

The addendum provides an optional steam cooling model that can be used for Emergency Core Cooling System (ECCS) Performance analyses to support the implementation of the CE 16 x 16 NGF fuel assembly design. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

Assumptions used for accident initiators and/or safety analysis acceptance criteria are not altered by the addition of these topical reports.

Safety Injection Tank Water Level and Containment Average Air Temperature

These values are used as inputs to the LBLOCA and SBLOCA analyses. The new limits ensure that the analyzed LBLOCA remain acceptable. The limits have no impact to the SBLOCA analysis results. The changes do not cause an increase in the probability of an accident or an increase in the dose consequences associated with a LBLOCA.

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

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

   Response: No.

Core Operating Limits Report (COLR)

The proposed change identifies changes in the codes used to confirm the values of selected cycle-specific reactor physics parameter limits. The proposed change allows the use of methods required for the implementation of CE 16 x 16 NGF. The proposed addition of the referenced topical reports has no impact on any plant configurations or on system performance that is relied upon to mitigate the consequences of an accident. The change to the COLR is administrative in nature and does not result in a change to the physical plant or to the modes of operation defined in the facility license.

WCAP-16500-P and Final Safety Evaluation

The proposed change adds Westinghouse topical report WCAP-16500-P, which describes the methods and models that will be used to evaluate the acceptability of CE 16 x 16 NGF at CE plants. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met. Prior to implementation of NGF, the new core design will be analyzed with applicable NRC staff approved codes and methods.

WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A

The proposed change allows the use of methods required for the implementation of Optimized ZIRLO™ clad fuel rods. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

WCAP-16523-P and Final Safety Evaluation

This topical report describes the departure from nucleate boiling correlations that will be used to account for the impact of the CE 16 x 16 NGF fuel assembly design. Entergy has demonstrated that the Limitations and Conditions associated with the SE will be met.

CENPD-387-P-A

The proposed addition of this topical report provides the departure from nucleate boiling (DNB) correlation that will be used to evaluate the DNB impact of non-mixing vane grid spans for CE 16 x 16 standard and NGF assemblies. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

CENPD-132, Supplement 4-P-A, Addendum 1-P and Final Safety Evaluation

The addendum provides an optional steam cooling model that can be used for ECCS Performance analyses to support the implementation of the CE 16 x 16 NGF fuel assembly design. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

Safety Injection Tank Water Level and Containment Average Air Temperature

The safety injection tank (SIT) system provides a passive means of adding a large quantity of borated water to the reactor core in the event of a LBLOCA. The SIT system serves no other purpose. Reducing the maximum volume will not create any new or different accidents.

The containment average air temperature ensures that the peak cladding temperature and cladding oxidation remain within limits during a LBLOCA. The change in the minimum allowable containment average temperature does not create any new or different accidents.

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

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

Response: No.

Core Operating Limits Report (COLR)

The addition of the following topical reports to the list of analytical methods referenced in the COLR is administrative in nature:

- WCAP-16500-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-16500-P, Revision 0, "CE [Combustion Engineering] 16x16 Next Generation Fuel [(NGF)] Core Reference Report"
- WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A
- WCAP-16523-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR), WCAP-16523-P, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes"
- CENPD-387-P-A
- CENPD-132, Supplement 4-P-A, Addendum 1-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) CENPD-132 Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE [Combustion Engineering] Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood"

Safety Injection Tank Water Level and Containment Average Air Temperature

The change to the allowable range for these two parameters does not reduce a margin of safety. The changes add to the margin of safety and provide assurance that the peak cladding temperature and cladding oxidation remain within limits.

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

Based on the above, Entergy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Entergy letter dated April 24, 2007 to the NRC, "License Amendment Request to Allow the Use of Optimized ZIRLO™ Fuel Rod Cladding," (W3F1-2007-0020)

2. Westinghouse Letter to the NRC, "Response to NRC's Request for Additional Information By the Office Nuclear Reactor Regulation Topical Report WCAP-16500-P, CE 16 x 16 Next Generation Fuel Core Reference Report (TAC No. MD0560, Proprietary/Non-proprietary)," LTR-NRC-06-66, November 29, 2006

3. CEN-356(V)-P-A Revision 01-P-A, "Modified Statistical Combination of Uncertainties," May 1988

4. Westinghouse Letter to the NRC, "Presentation Material on Audit Responses to Questions on Setpoints Supplement 1-P to WCAP-16500-P (TAC No. MD0560) (Proprietary/Non-proprietary)," LTR-NRC-07-20, April 5, 2007

5. NRC Letter to Westinghouse dated March 15, 2007, "Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR), WCAP-16523-P, 'Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes'"

6. NRC Letter to Westinghouse dated June 27, 2007, "Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) CENPD-132 Supplement 4-P-A, Addendum 1-P, 'Calculative Methods for the CE [Combustion Engineering] Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood'"

7. NRC Letter to Westinghouse dated July 30, 2007, "Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-16500-P, Revision 0, 'CE [Combustion Engineering] 16x16 Next Generation Fuel [(NGF)] Core Reference Report'"

Attachment 2

W3F1-2007-0037

Proposed Technical Specification Changes (mark-up)

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve open,

72.8%

b. A contained borated water volume of between 40% and ~~77.8%~~ level,

c. Between 2050 and 2900 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 670 psig.

APPLICABILITY: MODES 1, 2, 3*, and 4*.

ACTION: *MODES 1, 2, 3 and 4 with pressurizer pressure greater than or equal to 1750 psia.*

a. With one of the required safety injection tanks inoperable due to boron concentration not within limits, restore the boron concentration to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia within the following 6 hours.

b. With one of the required safety injection tanks inoperable due to inability to verify level or pressure, restore the tank to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia within the following 6 hours.

c. With one of the required safety injection tanks inoperable for reasons other than ACTION a or b, restore the tank to OPERABLE status within 24 hours, or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia within the following 6 hours.

* With pressurizer pressure greater than or equal to 1750 psia. When pressurizer pressure is less than 1750 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 235 psig and a maximum pressure of 670 psig, and a contained borated 72.8% water volume of between 61% and ~~77.8%~~ level. With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 235 psig and a maximum pressure of 670 psig, a boron concentration of between 2050 and 2900 ppm boron, and a contained borated water volume of between 39% and ~~77.8%~~ level. In MODE 4 with pressurizer pressure less than 392 psia (700 psia for remote shutdown from 72.8% LCP-43), the safety injection tanks may be isolated.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be $\geq$ 95 ~~90~~ °F* and $\leq$ 120 °F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

~~With the containment average air temperature outside the limits, restore the average air temperature to within the limits within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

INSERT 2

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at any three of the following locations and shall be determined at least once per 24 hours:

Location

a. Containment Fan Cooler No. 1A Air Intake

b. Containment Fan Cooler No. 1B Air Intake

c. Containment Fan Cooler No. 1C Air Intake

d. Containment Fan Cooler No. 1D Air Intake

* The minimum containment average air temperature limit is only applicable at greater than 70% RATED THERMAL POWER.

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT COLR (Continued)

6) "CESEC - Digital Simulation for a Combustion Engineering Nuclear Steam Supply System," (CE letter LD-82-001 and NRC SE to CE dated April 3, 1984). (Methodology for Specification 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating and group P CEA Insertion Limits, and 3.2.3 for Azimuthal Power Tilt).

7) "Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System," ENEAD-01-P. (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

8) "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A. (Methodology for Specification 3.2.1, Linear Heat Rate).

9) "Technical Description Manual for the CENTS Code," WCAP-15996-P-A. (Methodology for Specification 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating and group P CEA Insertion Limits, and 3.2.3 for Azimuthal Power Tilt).

10) "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," CENPD-132, Supplement 4-P-A. (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt and 3.2.7 for ASI).

11) "Implementation of ZIRLO Material Cladding in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

12) "Qualification of the PHOENIX-P/ANC Nuclear Design System For Pressurized Water Reactor Cores," WCAP-11596-P-A; "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A; and "ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery," WCAP-10965-P-A Addendum 1. (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

13) "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

14) "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," WCAP-16072-P-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

INSERT 1

6.9.1.11.2 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.11.3 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Insert 1

15) "CE 16 x 16 Next Generation Fuel Core Reference Report," WCAP-16500-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-16500-P, Revision 0, "CE [Combustion Engineering] 16x16 Next Generation Fuel [(NGF)] Core Reference Report," (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, 3.2.4.b, 3.2.4.c, and 3.2.4.d for DNBR Margin, and 3.2.7 for ASI).

16) "Optimized ZIRLO™," WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A, (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

17) "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," WCAP-16523-P-A and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR), WCAP-16523-P, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," (Methodology for Specification 3.2.4.b, 3.2.4.c, and 3.2.4.d for DNBR Margin).

18) "ABB Critical Heat Flux Correlations for PWR Fuel," CENPD-387-P-A (Methodology for Specification 3.2.4.b, 3.2.4.c, and 3.2.4.d for DNBR Margin and 3.2.7 for ASI).

19) "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood" CENPD-132, Supplement 4-P-A, Addendum 1-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) CENPD-132 Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE [Combustion Engineering] Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

Insert 2

a. If the minimum containment average air temperature is less than 95°F* but greater than or equal to 90°F*, then within 8 hours either restore containment air temperature to greater than 95°F or reduce the peak linear heat generation rate in accordance with the COLR. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
b. If maximum containment average air temperature is greater than allowable limit, then restore containment air temperature to within the limit within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Attachment 3

W3F1-2007-0037

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Additional growth data will be obtained from future Lead Test Assembly (LTA) exams ahead of the exposure achieved by batch implementation. This data will be provided to the NRC when it becomes available.			July 31, 2009 after LTA programs have burnups that bound current Waterford 3 burnup limits.
For the transition cycle, the DNB uncertainty addressable constants will not credit the DNB margin gain due to NGF, will not require application of the interim 6% margin penalty and will not require use of the 1/64 hypercube setpoints.	x		Start of Operating Cycle 16
The transition core LOCA evaluations and the revised full core LOCA analyses for ECCS Performance including the implementation of CE 16 x 16 NGF assemblies are being finalized and will be submitted to NRC for review.	x		
Since the optional steam cooling model as described in CENPD-132, Supplement 4-P-A, Addendum 1-P and Final Safety Evaluation is being used for Waterford 3 ECCS Performance Analyses, then a license amendment request will be submitted including the analyses and comparison graphical results needed to confirm the acceptability of the use of the optional steam cooling model.	x		