April 15, 2008

Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: REQUEST TO SUPPORT NEXT GENERATION FUEL; REVIEW AND APPROVAL OF REVISED EMERGENCY CORE COOLING SYSTEM (ECCS) PERFORMANCE ANALYSIS; AND REVIEW AND APPROVAL OF SUPPLEMENT TO THE ECCS PERFORMANCE ANALYSIS (TAC NOS. MD6954, MD6363, AND MD6954)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 214 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3 (Waterford 3). This amendment consists of: (1) changes to the Technical Specifications (TSs) in response to your application dated August 2, 2007, as supplemented by letters dated January 17, and March 10, 2008, and electronic mail dated March 24, 2008; (2) review and approval of your request submitted by letter dated August 9, 2007, as supplemented by letter dated January 21, 2008, the Waterford 3 revised Emergency Core Cooling System (ECCS) Performance Analysis that supports the implementation of Combustion Engineering (CE) 16x16 Next Generation Fuel (NGF) described in Westinghouse Topical Report (TR), WCAP-16500, "CE 16 x 16 Next Generation Fuel Core Reference Report"; and (3) review and approval of your request dated October 4, 2007, as supplemented by letter dated March 4, 2008, which provided a supplement to the ECCS performance analysis in support of NGF.

The proposed amendment changes the Waterford 3 TS 6.9.1.11.1, Core Operating Limits Report (COLR), TS 3.5.1, Safety Injection Tanks, and TS 3.6.1.5, Containment Air Temperature. The changes add new analytical methods and modify the containment average air temperature and safety injection tank level to support the implementation of NGF. The staff finds that the proposed changes to the Technical Specifications are acceptable.

The NRC staff has reviewed the ECCS performance analyses and concluded that the CE large break loss-of-coolant accident (LBLOCA) and small break loss-of-coolant accident (SBLOCA) analyses methodology, as described in CE TR, CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001, and CENPD-137, Supplement 2-A, Calculative Methods for the ABB CE Small break LOCA Evaluation Model, April 1998 (S2M Methodology), are acceptable for use by the licensee in demonstrating Waterford 3 compliance with the requirements of Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," of Title 10 of the *Code of Federal Regulations*. The Updated Final Safety Analysis Report will be changed to reflect the revised LOCA analyses.

The NRC staff also reviewed the supplementary analysis which addresses the limitations and conditions in the final NRC SE approving Westinghouse TR CENPD-132 Supplement 4-P-A, Addendum 1-P, "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 [inch per second] Core Reflood" dated June 27, 2007, and concluded that the analysis satisfactorily meets the limitations and conditions in the NRC SE for TR CENPD-132, Supplement 4-P-A, Addendum 1-P and is acceptable, and the results of the large-break LOCA analysis demonstrated that the use of the final NRC-approved version of the optional steam cooling model did not affect the results of the analysis of record (AOR) that used earlier version of the optional steam cooling model. The NRC staff, therefore, concluded that the AOR for Waterford 3 remained unchanged as a valid AOR.

A copy of our related SE analysis is enclosed.

The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures: 1. Amendment No. 214 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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Sincerely,
/RA/

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

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(*) SE input memo (**) See previous concurrence (***) No Legal Objection)

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	DSS/SRXB/BC	DSS/SNPB/BC	DSS/SCVB/BC	OGC	NRR/LPL4/BC
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ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 214
License No. NPF-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment by Entergy Operations, Inc. (EOI) dated August 2, 2007, as supplemented by letters dated January 17, and March 10, 2008, and electronic mail dated March 24, 2008; the request to review the revised emergency core cooling system (ECCS) performance analysis submitted by the letter dated August 9, 2007, as supplemented by letter dated January 21, 2008; and the request to review the supplement to the ECCS performance analysis submitted by letter dated October 4, 2007, as supplemented by letter dated March 4, 2008, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.2 of Facility Operating License No. NPF-38 is hereby amended to read as follows:

   2. <u>Technical Specifications and Environmental Protection Plan</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 214, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to startup following the spring 2008 refueling outage. Further, Facility Operating License No. NPF-38 is hereby amended to authorize a change to the Final Safety Analysis Report (FSAR) to reflect the revised loss-of-coolant accident analyses. The FSAR changes constitute a change in the analysis of record and will be a baseline for which future changes will be measured against in accordance with 10 CFR 50.46(a)(3). This action is required for the implementation of Next Generation Fuel as set forth in the license amendment application dated August 2, 2007, and evaluated in the safety evaluation dated April 15, 2008. The licensee shall update the FSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating
License No. NPF-38 and
Technical Specifications

Date of Issuance: April 15, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 214

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

Replace the following pages of the Facility Operating License and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
- 4 -	- 4 -

Technical Specifications

REMOVE	INSERT
3/4 5-1	3/4 5-1
3/4 6-13	3/4 6-13
6-20a	6-20a
6-20b	6-20b

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 214 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated August 2, 2007 (Reference 1), as supplemented by letters dated January 17 (Reference 4), and March 10, 2008 (Reference 5), and electronic mail dated March 24, 2008 (Reference 6), Entergy Operations, Inc. (Entergy, the licensee), requested changes to the Technical Specifications (TSs) for Waterford Steam Electric Station, Unit 3 (Waterford 3).

The proposed changes would revise the Waterford 3 TS 6.9.1.11.1, Core Operating Limits Report (COLR), TS 3.5.1, Safety Injection Tanks, and TS 3.6.1.5, Containment Air Temperature. The changes add new analytical methods and modify the containment average air temperature and safety injection tank (SIT) level to support the implementation of Next Generation Fuel (NGF). The licensee plans to implement the amendment during the spring 2008 refueling outage.

In addition, in Reference 1, Entergy had committed to address a limitation and condition in the final Nuclear Regulatory Commission (NRC) safety evaluation (SE) for the 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 [loss-of-coolant accident] Evaluation Model [EM] - Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood."

By letter dated August 9, 2007 (Reference 3), as supplemented by letter dated January 21, 2008 (Reference 7), Entergy submitted for the NRC review, the Waterford 3 revised Emergency Core Cooling System (ECCS) Performance Analysis that supports the implementation of CE 16x16 NGF described in Westinghouse TR WCAP-16500-P-A, “CE 16 x 16 Next Generation Fuel Core Reference Report,” per the commitment above. Attachment 1 of Reference 7 describes the Waterford 3 large-break LOCA (LBLOCA) analyses performed using the CENPD-132, Supplement 4-P-A methodology.

The NRC staff reviewed the licensee’s demonstration evaluations of the ECCS performance, done in accordance with the CENPD-132, Supplement 4-P-A, Addendum 1-P methodology for Waterford 3 operating at its currently licensed core power of 3716 megaWatts thermal (MWt).

The specific Waterford 3 analyses were performed to demonstrate the suitability of the CE methodology for application to Waterford 3. Also, the LBLOCA analyses, discussed herein, will be acceptable and specifically applicable to Waterford 3 operated with the fuel(s) identified in the Table 1 that follows. The CE LBLOCA analyses for Waterford 3 were conducted assuming that the plant uses cores containing NGF with Optimized ZIRLO™ clad uranium oxide fuel assemblies.

Finally, by letter dated October 4, 2007 (Reference 11), and supplemented by letter dated March 4, 2008 (Reference 12), the licensee provided a supplement to the ECCS performance analysis submittal for the NRC staff to review and approve.

The revised LBLOCA analysis was performed using a new optional steam cooling model improvement in the 1999 Westinghouse ECCS performance evaluation model (EM) for CE plants (Reference 13). The licensee completed the LBLOCA analysis of record (AOR) in Reference 3 before it received the NRC's final safety evaluation report (SER, Reference 8) approving the new optional steam cooling model. The final NRC SER imposed several limitations and conditions on the use of the cooling model. In addressing the SER restrictions and conditions, the licensee provided a supplementary analysis to (1) document the performance of the optional steam cooling model in its final approved form, and (2) demonstrate compliance with the final SER limitations and conditions.

Also, the licensee presented the results of the LBLOCA analysis to demonstrate that the use of the final NRC-approved version of the optional steam cooling model did not affect the results of the AOR (Reference 3) that used an earlier version of the optional steam cooling model, and proposed that the AOR in Reference 3 for Waterford 3 remained as a valid AOR.

The supplemental letters dated January 17, and March 10, 2008, and electronic mail dated March 24, 2008, for changes to the TSs; the supplemental letter dated January 21, 2008, for review and approval of the revised ECCS performance analysis; and the supplemental letter dated March 4, 2008, for review and approval of the supplement to the ECCS performance analysis, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 11, 2007 (72 FR 51858).

Section 1 of this safety evaluation addresses the requested changes to the TSs, Section 2 addresses the review and approval of the revised ECCS Performance Analysis that supports the implementation of CE 16x16 NGF, and Section 3 addresses the review and approval of the supplement to the ECCS performance analysis

2.0 REGULATORY EVALUATION

10 CFR Part 50 includes the NRC's requirement that TSs shall be included by applicants for a license authorizing operation of a production or utilization facility. 10 CFR 50.36(d) requires that TSs include items in five specific categories related to station operation. These categories are: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The proposed change to TSs is related to the safety limits, limiting safety system settings, and limiting controls settings category.

2.1 Technical Specification Changes

2.1.1 Core Operating Limits Report (COLR), TS 6.9.1.11.1

Regulatory guidance for the review of fuel rod cladding materials and fuel system designs and adherence to applicable General Design Criteria (GDC) is provided in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 4.2, "Fuel System Design." In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences,
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- The number of fuel rod failures is not underestimated for postulated accidents, and
- Coolability is always maintained.

In addition to licensed reload methodologies, an approved mechanical design methodology is utilized to demonstrate compliance with SRP 4.2 fuel design criteria. The NRC staff has previously reviewed and approved the CE 16x16 NGF fuel assembly design for application in CE plant designs (Reference 2).

The proposed TS changes are evaluated to ensure continued compliance with the requirements of 10 CFR 50.36(d)(2)(ii), listed below.

(ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
(B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
(C) Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
(D) Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Compliance with this regulation requires a licensee to maintain a list of approved analytical methods (used to establish potentially cycle-specific core operating limits, per NRC Generic Letter (GL) 88-16).

2.1.2 Containment Air Temperature, TS 3.6.1.5

The following design criteria from Appendix A of 10 CFR Part 50, apply:

(1) GDC 16 as it relates to the containment and associated systems establishing a leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident requires;
(2) GDC 38 as it relates to the containment heat removal system safety function which shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident (LOCA) and to maintain them at acceptably low levels; and
(3) GDC 50 as it relates to the containment heat removal system which shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

Appendix K of 10 CFR 50, Section I.D.2, which requires that the containment pressure used for evaluating the effectiveness of emergency core cooling shall not exceed a pressure calculated conservatively for this purpose.

The NRC staff used the following sections of NUREG-0800 for this review:

- 6.2.1, “Containment Functional Design,”
- 6.2.1.1.A, “PWR [Pressurized-Water Reactor] Dry Containments, Including Subatmospheric Containments,”
- 6.2.1.5, “Minimum Containment Pressure Analysis for Emergency Core Cooling System [ECCS] Performance Capability Studies,” and
- 6.2.2, ”Containment Heat Removal Systems.”

2.1.3 Safety Injection Tanks, TS 3.5.1

The regulatory requirements that the NRC staff considered in its review of the specific changes to TS 3.5.1 are in 10 CFR 50.46(a)(1)(i), which requires that the ECCS performance analysis must conform to the ECCS acceptance criteria of 10 CFR 50.46(b).

2.2 Review and Approval of Emergency Core Cooling System Performance Analysis

The LBLOCA analyses were performed to demonstrate that the ECCS design would provide sufficient ECCS flow to transfer the heat from the reactor core following an LBLOCA at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) the clad metal-water reaction would be limited to less than the amounts that would compromise cladding ductility and result in excessive hydrogen generation. The NRC staff reviewed the analyses (References 3 and 7) to assure that the safety functions could be accomplished with appropriate consideration of single failure, containment capabilities and loss of onsite or offsite electric power (i.e., assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available, with offsite electric power available). The NRC staff used the acceptance criteria for ECCS performance provided in 10 CFR 50.46, in assessing the acceptability of the CENPD-132, Supplement 4-P-A, Addendum 1-P LBLOCA methodology for Waterford 3.

In its assessment of the acceptability of the methodology for Waterford 3, the NRC staff also reviewed the limitations and conditions stated in its SE (Reference 8) supporting general approval of the CENPD-132, Supplement 4-P-A, Addendum 1-P LBLOCA methodology and the range of parameters described in the CENPD-132, Supplement 4-P-A, Addendum 1-P LBLOCA methodology.

2.3 Review and Approval of Supplementary Emergency Core Cooling System Performance Analysis

10 CFR 50.46 specifies that the performance of an ECCS shall be calculated in accordance with an acceptable EM for a limiting LOCA to meet the following acceptance criteria: the PCT does not exceed 2200 °F; the maximum cladding oxidation does not exceed 17 percent of the total cladding thickness; the maximum metal-water reaction does not exceed 1 percent of the total amount of metal in the core; and the core geometry remains amenable to long-term cooling.

Condition 4 of the NRC SER approving the new optional steam cooling model (Reference 8) stated, in part, that "...the licensee should provide the results of the evaluations 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 PCT, 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 optional steam cooling model."

The NRC staff's review of the supplementary LBLOCA analysis in Reference 11 and the licensee's response to the request for additional information (RAI) in Reference 12 was to assure that (1) the licensee used the EM in compliance with the restrictions and conditions the SER approving the new optional steam cooling model, (2) the results of the analysis were within the applicable ECCS performance criteria set forth in 10 CFR Part 50.46. The review also would determine the acceptability of the proposed AOR in Reference 8 as a valid AOR.

3.0 <u>TECHNICAL EVALUATION</u>

3.1 Technical Specification Changes

3.1.1 Proposed Change to TS 6.9.1.11.1

The proposed change to TS 6.9.1.11.1, Core Operating Limits Report (COLR), involves adding new analytical methods related to NGF which will be used to determine the core operating limits. The proposed changes are provided in Attachment 2 of Reference 1 with the justification provided in Attachment 1 of Reference 1. The new analytical methods being added to the COLR, listed below, have been previously reviewed and approved by the NRC.

- "CE 16x16 Next Generation Fuel Core Reference Report," WCAP-16500-P, Rev. 0, and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-16500-P, Revision 0, "CE 16x16 Next Generation Fuel [(NGF)] Core Reference Report," July 30, 2007.

- WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, “Optimized ZIRLO™,” February 2003.
- “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,” March 15, 2007.
- “ABB Critical Heat Flux Correlations for PWR Fuel,” CENPD-387-P-A., Rev. 000, May 2000.
- “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 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,” June 27, 2007.

Attachment 1 of Reference 1 identifies and dispositions all of the SE conditions and limitations within each of the licensed topical reports (LTRs) being added to COLR. Part of the disposition includes regulatory commitments which are highlighted in Attachment 3 of Reference 1. The staff reviewed the SE conditions and limitations and finds that the licensee adequately addressed each one of them.

The staff verified that each of the LTRs being added to TS 6.9.1.11.1 are applicable to the CE-designed Waterford 3 reactor core. Based upon compliance with SE limitations and conditions and the regulatory commitments identified in Attachment 3 of Reference 1, the staff finds the proposed changes to TS 6.9.1.11.1 acceptable.

The NRC staff also identified that for a particular cycle-specific core operating limit, many approved analytical methods are used. According to the Generic Letter 88-16 (GL 88-16) guidance, TS 6.9.1.11.1 should list only the main approved methods currently used to support the cycle-specific core operating limit. Therefore, the NRC staff requested the licensee to identify the main method used to reflect the GL 88-16 guidance to minimize the number of the approved methods listed in TS 6.9.1.11.1. In response to the NRC staff request, the licensee has committed (Reference 5) to reduce the number of references consistent with the guidance specified in GL 88-16 within 12 months following NRC issuance of the approved amendment for the current requested changes to TS 6.9.1.11.1. The commitment is acceptable.

Summary

The addition of five new analytical methods used to determine the core operating limits for implementation of NGF are acceptable because: (1) the proposed five analytical methods are approved methodologies with licensee’s analyses for their limitations and conditions applied to Waterford 3 operation; and (2) the licensee also identified a specific approved methodology to be used for calculating a cycle-specific core operating limits specified in TS 6.9.1.11.1. The licensee’s commitment to maintain within the COLR only the current methods used to determine core operating limits is acceptable to the NRC staff.

3.1.2 Proposed Change to TS 3.6.1.5

The Waterford 3 containment structure is comprised of a steel containment vessel surrounded by reinforced concrete.

The licensee proposes to revise the minimum containment operating temperature limit from 90 degrees Fahrenheit (°F) to 95 °F and states that the only accident analysis affected due to this change is the ECCS performance analysis for LBLOCA which uses the minimum allowed containment temperature as an initial condition for conservatism in determining the peak cladding temperature (PCT). By Reference 3, the licensee submitted a revised ECCS performance analysis for an LBLOCA which used the minimum allowed temperature of 95 °F as the containment initial condition. For this analysis, the licensee used computer code COMPERC-II which is the same as used for the current licensing basis analysis. The staff reviewed the revised transient containment pressure and temperature, and sump water temperature curves given in Reference 3, Figures 5-6, 5-21, and 5-22, respectively, and compared these curves with the current licensing basis curves given in updated Final Safety Analysis Report (FSAR) Figures 6.2-31a, 6.2-31b, and 6.2-31c and found differences between the Reference 3 figures and FSAR figures very small and thus negligible.

Summary

The licensee proposes to revise the containment minimum operating temperature limit from 90 °F to 95 °F. The staff determined that the proposed changes meet the requirements of 10 CFR Part 50 Appendix A, (a) GDC 16 because the change does not affect the current licensing basis peak containment pressure and temperature analysis and, therefore, the primary containment design conditions important to safety are not exceeded, (b) GDC 38 because the change does not affect the current licensing basis containment heat removal system which maintains the containment pressure and temperature below their design limits following a design-basis accident, (3) GDC 50 because there is no effect on the primary containment heat removal system which is designed so that the primary containment structure and its internal compartments can accommodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from the design-basis accident.

Therefore, the NRC staff considers the transient containment pressure and temperature, and sump water temperature curves obtained from the revised "Minimum Containment Pressure Analysis for Performance Capability Studies on the Emergency Core Cooling System," as acceptable, when compared with the FSAR Section 6.2 values.

3.1. 3 Proposed Change to TS 3.5.1

The proposed change to TS 3.5.1, Safety Injection Tanks, reduces the maximum allowable SIT water level from 77.8 percent to 72.8 percent.

The SITs are used to flood the core with borated water following depressurization as a result of a LOCA and/or Main Steam Line Break. A lower maximum SIT water level results in a larger initial gas volume in the SIT. With a larger initial gas volume, as the SIT injects during the LBLOCA, the higher gas volume in the SIT provides a slightly greater driving force for SIT water to be injected into the Reactor Coolant System (RCS) as the RCS pressure lowers, thereby increasing the reactor vessel reflood rate. 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 the PCT for an LBLOCA.

The licensee provides the technical basis to justify that the change in SIT level (from 77.8 percent to 72.8 percent) still meets the conservative maximum allowable SIT level based on the results of the ECCS Performance Analysis (Reference 2) given in Table 5.1 of Reference 1. The staff has reviewed the justification for the licensee's technical basis to support this proposed TS change and found it acceptable because an approved methodology has been used and the analysis produces a conservative result.

Summary

The staff concludes that lowering the maximum SIT level from 77.8 percent to 72.8 percent is acceptable because the proposed change is based on the result of the LBLOCA analysis using an approved methodology.

In addition to the proposed changes above, the licensee identified administrative changes on TS pages 6-20a and 6-20b, such as the text that is currently at the bottom of page 6-20a associated with TS 6.9.1.11.2 and TS 6.9.1.11.3 will be moved to page 6-20b as will some of the above-listed references. The staff finds these administrative changes acceptable.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the regulatory commitments are best provided by the licensee's administrative processes, including its commitment management program. The regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

3.2 Review and Approval of Emergency Core Cooling System Performance Analysis

In Reference 7, the licensee provided the results for the Waterford 3 LBLOCA analyses operating at a power of 3735 MWt (while the rated core power is 3716 MWt, analyses were performed at 3735 MWt to account for a 0.5 percent power measurement uncertainty) performed in accordance with the CE LBLOCA methodology. The licensee's results for the calculated PCTs, the maximum cladding oxidations (local), and the maximum core-wide cladding oxidations for Waterford 3 are provided in the Table 1 along with the acceptance criteria of 10 CFR 50.46(b).

The concern with core-wide oxidation relates to the amount of hydrogen generated during a LOCA. Because hydrogen that may have been generated pre-LOCA (during normal operation) will be removed from the reactor coolant system throughout the operating cycle, the NRC staff noted that pre-existing oxidation does not contribute to the amount of hydrogen generated post-LOCA and, therefore, it does not need to be addressed further when determining whether the calculated total core-wide hydrogen meets the 1.0 percent criterion of 10 CFR 50.46(b)(3).

As discussed above, Entergy had Westinghouse conduct the LBLOCA analyses for Waterford 3 operating at a power level of 3735 MWt using an NRC-approved CE LBLOCA analysis methodology. The NRC staff concluded that the results of these analyses indicated compliance with 10 CFR 50.46(b)(1) through (b)(3) for power levels of up to 3735 MWt. Meeting these criteria provides reasonable assurance that at the current licensed power level, the Waterford 3 core will be amenable to cooling as required by 10 CFR 50.46(b)(4). The capability of Waterford 3 to satisfy the long-term cooling requirements of 10 CFR 50.46(b)(5) is unaffected by the proposed LAR. The LBLOCA analysis methodology used to perform the analyses discussed above continues to be acceptable and suitable for inclusion in the Waterford 3 Core Operating Limits Report (COLR).

TABLE 1
LARGE BREAK LOCA ANALYSIS RESULTS
CENPD-132, SUPPLEMENT 4-P-A, ADDENDUM 1-P

Parameter	Waterford 3 Results*	10 CFR 50.46 Limits
Cladding Material	Zirlo	(Cylindrical) Zircaloy or Zirlo
Peak Clad Temperature	2166 ^{o}F	2200 ^{o}F (10 CFR 50.46(b)(1))
Maximum Local Oxidation	16.9 percent	17.0 percent (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	<1 percent	1.0 percent (10 CFR 50.46(b)(3))

* for a double ended guillotine break at the reactor coolant pump discharge leg.

The small-break LOCA (SBLOCA) methodology used for the analyses was found to apply to all conventional CE PWR designs in the NRC generic SE of CENPD-137, Supplement 2 P-A (S2M) methodology (Reference 9). Therefore, the SBLOCA methodology described in CENPD-137, Supplement 2-P-A, is acceptable for application to Waterford 3, which is a PWR designed by CE, and for inclusion in the Waterford 3 TSs. The above-listed TS Reference the Supplement 2 version (referred to as the S2M or Supplement 2 Model) was identified in the licensee's submittal as a TS reference.

The licensee's results from the SBLOCA analyses, for the calculated PCTs, the maximum cladding oxidations (local), and the maximum core-wide cladding oxidations for Waterford are provided in Table 2, shown below, along with the acceptance criteria of 10 CFR 50.46(b).

TABLE 2
SMALL-BREAK LOCA ANALYSIS RESULTS CENPD-137, SUPPLEMENT 2-A

Parameter	Waterford 3 Results*	10 CFR 50.46 Limits
Limiting Break Size/Location	0.05ft^2/PD	Not Applicable
Cladding Material	Zirlo	(Cylindrical) Zircaloy or Zirlo
Peak Clad Temperature	1944 ^{o}F	2200 ^{o}F (10 CFR 50.46(b)(1))
Maximum Local Oxidation	14.3 percent	17 percent (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	<1 percent	1.0 percent (10 CFR 50.46(b)(3))

* for a break at the pump discharge.

Therefore, the NRC staff concludes that the S2M methodology is applicable to Waterford 3 and that the limitations and conditions of the NRC's SE approving the S2M methodology (Reference 10) were satisfied for the Waterford 3 rated power level and the fuels discussed above. The staff also concludes that the S2M applies to Waterford 3 and, therefore, the proposed addition of S2M to the Waterford 3 TS is acceptable.

In Reference 7, the licensee stated, "Waterford 3 and its vendor, Westinghouse Electric Company LLC, continue to have ongoing processes, which assure that LOCA analysis input values bound the as-operated plant values for those parameters." The NRC staff finds that this statement, along with the generic acceptance of the CE LBLOCA analysis methodology, provides assurance that the CE CENPD-132, Supplement 4-P-A, Addendum 1-P LBLOCA analysis methodology and LBLOCA analyses are applicable to Waterford 3, operated at its current licensed power level.

The staff also finds that S2M report used at Waterford 3 and the version is suitable for listing in the COLR for the Waterford 3 plant, consistent with guidance provided in NUREG-1432.

Summary

Based on this review, the NRC staff concludes that the CE LBLOCA and SBLOCA analyses methodologies, as described in TRs CENPD-132-P-A and CENPD-137-P-A, respectively, are acceptable for use by Entergy in demonstrating that Waterford 3 complies with the requirements of 10 CFR 50.46(b), because the Waterford 3 plant design is among the designs for which application of the CE LBLOCA and SBLOCA methodologies was approved by the NRC (References 8 and 10).

The NRC staff's review of the acceptability of the CE LBLOCA and SBLOCA methodologies for Waterford 3 focused on assuring that the licensee and its vendor have processes to assure that specific input parameters or bounding values are used to conduct the Waterford 3 LBLOCA and SBLOCA analyses, that the analyses will be conducted within the conditions and limitations stated in the NRC-approved CE LBLOCA and SBLOCA methodologies, and that the results will satisfy the requirements of 10 CFR 50.46(b) for Waterford 3 operating at its present licensed power.

This SE also documents the NRC staff review and acceptance of the CE LBLOCA and SBLOCA analysis methodology for application to Waterford 3, for inclusion in the Waterford 3 TS and COLR, and of the specific LBLOCA and SBLOCA analyses discussed above that were performed with NRC-approved CE LBLOCA and SBLOCA methodologies for Waterford 3 operated at powers up to its licensed power level of 3716 MWt.

3.3 Review and Approval of Supplementary Emergency Core Cooling System Performance Analysis

3.3.1 1999 EM Optional Steam Cooling Model

The licensee performed supplementary LBLOCA analysis using the 1999 NRC approved EM: CEFLASH-4A/FII for the blowdown analysis; COMPERC-II/LB for the reflood analysis; STRIKIN-II for the hot rod heatup analysis; and COMZIRC for the corewide oxidation calculation.

The 1999 EM included an optional steam cooling model that was documented in Reference 13 and approved by the NRC based on the bases in its SER documented in Reference 8 with restrictions and conditions specified in Section 4.0 of the SER.

Conditions 1 and 2 of the SER ensure that the application of the selected spacer grid rewet temperature and the optional steam cooling model in Reference 13 was limited to the CE 16X16 NGF fuel. Since the licensee applied the model to its implementation of the CE 16x16 NGF fuel assemblies in Waterford 3, Conditions 1 and 2 were satisfied.

Condition 3 of the SER ensures that the results of the grid model enhancement cannot result in the use of a heat transfer coefficient greater than FLECHT. The licensee indicated (Reference 1) that the PARCH module of the STRIKIN-II used for the revised LBLOCA analysis contained a specific algorithm to insure that the Condition 3 constraint on the use of the FLECHT upper-bound heat transfer coefficient. Therefore, Condition 3 was met.

Conditions 4 specified that when a licensee wanted to use the optional steam cooling model, a LAR should be submitted including the analyses performed to determine its applicability to specific fuel design being evaluated, using the approved model discussed in Sections 3.3.1, 3.3.2, and 3.3.3 of the SER for the selected spacer grid rewet temperature, spacer grid heat transfer coefficient, and spacer grid geometry and flow blockage, respectively. The licensee indicated that (Reference 11) its revised LBLOCA analysis used the model that included (1) the approved spacer grid rewet temperature criterion required by the final SER and (2) final formulation of calculated parameters required by the final SER. With meeting Condition 5 discussed below and an acceptable LBLOCA analysis in Section 3.3.3.1 of this evaluation report, the Condition 4 limitations were satisfactorily met.

Condition 5 specified that the optional steam cooling model should be used within the applicable range of flow blockage and Reynolds number discussed in Section 3.3.3 of the SER. The licensee indicated (Refs. 11 and 1) that the PARCH module of the STRIKIN-II hot rod heatup computer code contained specific computational constrains to print a warning and diagnostic output message to alert the user if the calculation is found to be outside the range of applicability for flow blockage and Reynolds number. These PARCH features provided reasonable assurance that the Condition 5 limitations were met.

3.3.2 LBLOCA Analysis

The licensee analyzed the following two cases identified in the current AOR as the limiting cases:

1. the limiting PCT case – 1.0 double-ended guillotine break in the reactor coolant pump discharge leg (DEG/PD) based on the UO_2 fuel type with optimized ZIRLO cladding at a burnup 32 gigaWatt day per metric ton unit (GWD/MTU), and

2. the limiting local cladding oxidation percentage case – 1.0 DEG/PD based on the UO_2 fuel type with optimized ZIRLO cladding at a burnup 0.5 GWD/MTU.

Similar to the AOR, both cases assumed no failure in the ECCS as the worst single failure event. The safety injection tank initial conditions and refueling water storage initial conditions that led to the most limiting ECCS performance results in the AOR were used in the supplementary LOCA analysis.

During the course of the review, the NRC staff requested the licensee to justify that the AOR limiting cases remained to be the limiting cases when the final NRC-approved optional steam cooling model was used. In response (Reference 12), the licensee indicated that the limiting cases for PCT and local cladding oxidation percentage identified in the Waterford 3 AOR (Reference 3), using the earlier version of the optional steam cooling model, were not affected by the minor differences with the final optional steam cooling model. As demonstrated by the comparisons provided in the Reference 3 and in the response to the RAI (Reference 12), the tabulation of key parameters showed that there was no effect of using the final optional steam cooling model on the calculated results. The calculational results were the same regardless of the boundary conditions for any given case because of the nature of the differences between the final optional steam cooling model and the earlier model.

For application of the LBLOCA EM to Waterford 3, the results of sensitivity study analysis in the AOR showed that the limiting PCT condition occurred exclusively on the node below the rupture node elevation. This was based on the limiting core reflood rates calculated for the assumed ECCS equipment boundary conditions. The occurrence of this limiting condition below the rupture node elevation for PCT was not affected by the Appendix K reflood steam cooling model requirements or the optional steam cooling model, which were only applied to the rupture node elevation and above in the Westinghouse 1999 EM for an LBLOCA. Therefore, limiting PCT cases were not affected by the use of the optional steam cooling model in either its final version or the earlier version.

Also, the determination of the limiting cases for LBLOCA was dominated by the selected boundary conditions. The specified break size and location and the assumed ECCS equipment specifications affected the calculated system responses for the blowdown period by CEFLASH-4A and for the reflood period by COMPERC-II. Neither of these two licensed computer codes or their calculated results was affected by the optional steam cooling model. Therefore, the reactor coolant system blowdown and reflood thermal-hydraulic results, which were transferred to the hot rod heatup calculation as boundary conditions, were not affected by the optional steam cooling model.

In addition, the optional steam cooling model was used in STRIKIN-II for calculating hot rod heatup at and above the rupture node elevation when the core reflood rate was less than 1 in/sec. The specified fuel design type and the time-in-life controlled the selection of the initial conditions for rod internal pressure and fuel stored energy for the hot rod heatup calculation, which ultimately led to the determination of the limiting cases. The time of less than 1 in/sec core reflood occurred late in the reflood process. The effects of initial fuel stored energy and rod internal pressure had already influenced the hot rod heatup calculation through the blowdown period and early reflood period before the time of 1 in/sec core reflood. Therefore, the determination of the limiting cases as influenced by fuel type and time-in-life was not affected by the optional steam cooling model.

Furthermore, cladding rupture was calculated to occur before the time of less than 1 in/sec core reflood in the AOR spectrum of cases. As stated above, the PCT occurred below the rupture node elevation and was not affected by the steam cooling model, which was applied only to the rupture node and above when the core reflood rate was less than 1 in/sec. Therefore, the determination of the limiting PCT cases as influenced by cladding rupture was not affected by the optional steam cooling model.

For the limiting local cladding oxidation percentage case, the licensee indicated that the steam cooling heat transfer coefficient was constrained to be no better than the heat transfer coefficient calculated with the FLECHT correlation. The FLECHT heat transfer coefficients were dependent on the COMPERC-calculated reflood rates and were not affected by the optional steam cooling model. Referring to Figure 3.3-5 of the Reference 8, the FLECHT correlation was the source of the steam cooling heat transfer coefficient on the rupture node where peak local oxidation occurred until near the end of the transient at roughly 470 seconds. The final coding changes altered the spacer grid steam cooling calculation slightly, but the value of the steam cooling heat transfer coefficient utilized in the rupture node calculation was not changed since it was limited by the FLECHT value for almost the entire reflood period. The only effect of the change in the optional steam cooling model would be seen as a slightly earlier time when the grid heat transfer value dropped below the FLECHT value at or near the end of the transient (around 470 seconds), which was after the cladding temperature had turned around and after the oxidation calculation had reached its limit. Therefore, the determination of the limiting local cladding oxidation percentage cases was not affected by the optional steam cooling model.

3.3.3 Results of LBLOCA Analysis

3.3.3.1 The LBLOCA Analysis with and without the Optional Steam Cooling Model

The licensee compared the results of the supplementary LBLOCA analysis performed with and without the option steam cooling model to quantify the performance of the model and show its effect on the calculated results. The licensee's analysis indicated that use of the optional steam cooling model had no effect on the calculated PCT for the implementation of CE16x16 NGF at Waterford 3, since the PCT occurred below the rupture node that was not subjected to any steam cooling heat transfer limitation.

The use of the optional steam cooling model would improve the heat transfer coefficient on the fuel rod rupture node and provide margin to the peak local oxidation criterion. As demonstrated by the results of the supplementary LBLOCA analysis, Case 2 discussed in Section 3.2 (Section 3), the limiting local cladding oxidation case indicated that the peak local cladding without the steam cooling model exceeded the ECCS acceptance criterion of 17 percent. Also, the core-wide oxidation percentage exceeded the ECCS acceptance criterion of 1 percent. When the optional steam cooling model was included in the analysis, the results of the analysis for the same Case 2 demonstrated that the peak local cladding oxidation (16.9 percent) and core-wide oxidation (0.988 percent) were within respective ECCS acceptance criteria of 17 percent and 1 percent.

The licensee also provided supplementary graphical results that included PCT, local oxidation, and steam cooling flow rates in References 11 and 12. A comparison of the graphical results with and without the optional steam cooling model showed that the results were consistent with the range and order of magnitude of the model-difference effect discussed in the NRC approval process for the optional steam cooling model. Therefore, the staff concluded that the graphical results provided by the licensee satisfied the second part of Condition 4 in the NRC SER that stated that: "the licensee should provide the results of the evaluations 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 PCT, local oxidation, and steam cooling flow rates should be included in the submittal."

3.3.3.2 Results for the Limiting PCT and Local Cladding Oxidation Cases with the Final Approved Optional Steam Cooling Model

The licensee provided in Table 3.4-1 (Reference 11) and the table to RAI-1 (Reference 12) comparisons of results between the limiting PCT and local oxidation cases from the AOR (Reference 3) using earlier version of the model without implementation of the SER conditions in Reference 8 and the cases using the final approved optional steam model (Reference 8). The comparisons showed that there was no impact on key results between the two versions of the optional steam cooling model. The results of both supplementary analysis in Reference 11 and the AOR in Reference 3 demonstrated that the PCT does not exceed 2200 °F; the maximum cladding oxidation does not exceed 17 percent of the total cladding thickness; the maximum metal-water reaction does not exceed 1 percent of the total amount of metal in the core; and the core geometry remains amenable to long-term cooling. Therefore, the NRC staff concluded that the proposed AOR for Waterford 3 presented in Reference 3 remained as a valid AOR.

Summary

Based on its review, the NRC staff found that (1) the supplementary LBLOCA analysis were performed based on the final NRC-approved optional steam cooling model in compliance with the restrictions and conditions in the SER approving the model, and (2) the results of the supplementary LBLOCA analysis showed compliance with 10 CFR 50.46 acceptance criteria: the PCT does not exceed 2200 °F; the maximum cladding oxidation does not exceed 17 percent of the total cladding thickness; the maximum metal-water reaction does not exceed 1 percent of the total amount of metal in the core; and the core geometry remains amenable to long-term cooling. Therefore, the NRC staff concluded that the supplementary LBLOCA analysis was acceptable. Also, the results of the LBLOCA analysis demonstrated that the use of the final NRC-approved version of the optional steam cooling model did not affect the results of the AOR that used earlier version of the option steam cooling model. The NRC staff, therefore, concluded that the proposed AOR for Waterford 3 presented in Reference 3 remained unchanged as a valid AOR.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published September 11, 2007 (72 FR 51858). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter dated August 2, 2007, from Kevin T. Walsh (Entergy), to US NRC, “License Amendment Request NPF-38-271 to Support Next Generation Fuel” ((Agencywide Documents Access and Management System (ADAMS) Accession No. ML072180042).

2. WCAP-16500-P-A, “CE 16x16 Next Generation Fuel Core Reference Report” (ADAMS Accession No. ML072500331).

3. Letter dated August 9, 2007, from Kevin T. Walsh, Vice President, Operations, Entergy Nuclear South, Entergy Operations, Inc, to US NRC, Emergency Core Cooling System Performance Analysis, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38 (ADAMS Accession Number ML072250389).

4. Letter dated January 17, 2008, from Kimberly S. Cook (Entergy), to US NRC, “Supplement to Amendment Request NPF-38-271 to Support Next Generation Fuel” (ADAMS Accession No. ML080220070).

5. Letter dated March 10, 2008, from Kimberly S. Cook (Entergy), to US NRC, “Supplement 2 RAI Response to Amendment Request NPF-38-271 to Support Next Generation Fuel” (ADAMS Accession No. ML080720667).

6. Email dated March 24, 2008, from R. Williams (Entergy), to N. Kalyanam (NRC), “Ongoing Processes” (ADAMS Accession No. ML080840484).

7. Letter from Entergy (Kimberly S. Cook) to U.S. Nuclear Regulatory Commission dated January 21, 2008, “Request for Additional Information Related for Emergency Cooling System Performance Analysis” (ADAMS Accession No. ML080230557).

8. Letter from H. K. Nieh (NRC) J. A. Gresham (Westinghouse), “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 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’ (TAC No. MD2161),” dated June 27, 2007.

9. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB [Asea Brown Boveri] CE Small Break LOCA Evaluation Model," April 1998.

10. NRC Safety Evaluation related to, and attached to, the S2M Methodology documentation, CENPD-137, Supplement 2-P-A, April 1998.

11. Letter from R. J. Murillo (Entergy) to NRC, "Supplement to the ECCS Performance Analysis Submit in Support of Next Generation Fuel in Waterford 3 – 1999 EM Optional Steam Cooling Model Justification, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," dated October 4, 2007, ADAMS Accession No. ML072820400.

12. Letter from R. J. Murillo (Entergy) to NRC, "Response to Request for Additional Information for Supplement to the ECCS performance analysis Submittal in Support of next generation Fuel in Waterford 3 – 1999 EM Optional Steam Cooling Model Justification," dated March 4, 2008.

13. Westinghouse Topical Report, CENPD-132, Supplement 4-P-A, Addendum 1-P, "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," dated May 2006.

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