October 31, 2003

Mr. Joseph E. Venable
Vice President Operations
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: ADDITION OF TOPICAL REPORT ENTITLED "FUEL ROD MAXIMUM ALLOWABLE GAS PRESSURE," CEN-372-P-A, TO THE LIST OF ANALYTICAL METHODS IN TECHNICAL SPECIFICATION 6.9.1.11.1 TO DETERMINE THE CORE OPERATING LIMITS (TAC NO. MB6964)

Dear Mr. Venable:

The Commission has issued the enclosed Amendment No. 191 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3 (Waterford 3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 16, 2002, as supplemented by letters dated July 30, and September 29, 2003.

The amendment adds topical report CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," dated May 1990, to the list of topical reports in TS 6.9.1.11.1, used to determine the Waterford 3 core operating limits. In addition, the amendment approves the deletion of applicable dates and revision numbers for CEN-372-P-A and other topical reports listed in TS 6.9.1.11.1. The latter change is consistent with approved TS Task Force (TSTF) traveler TSTF-363.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

N. Kalyanam, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures: 1. Amendment No. 191 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
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 December 16, 2002, as supplemented by letters dated July 30, and September 29, 2003, complies 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. Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 191, 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 within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 31, 2003

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 191</u>

<u>TO FACILITY OPERATING LICENSE NO. NPF-38</u>

<u>DOCKET NO. 50-382</u>

Replace the following pages of the 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.

Remove	Insert
6-20	6-20
6-20a	6-20a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 191 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 December 16, 2002, as supplemented by letters dated July 30, and September 29, 2003, Entergy Operations, Inc. (Entergy or the licensee), requested changes to the Technical Specifications (TSs) for Waterford Steam Electric Station, Unit 3 (Waterford 3).

The amendment would add the Combustion Engineering (CE) topical report (TR) CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990, to the list of references in TS 6.9.1.11.1, used to determine the Waterford 3 core operating limits. In addition, the amendment would approve the deletion of applicable dates and revision numbers for CEN-372-P-A and other TRs listed in TS 6.9.1.11.1, Items 1 through 8. This change is consistent with TS Task Force (TSTF) Item TSTF-363.

The supplemental letters dated July 30, and September 29, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (68 FR 5673, dated February 4, 2003).

2.0 REGULATORY EVALUATION

The staff finds that the licensee, in Section 5.0 of its submittal, identified the applicable regulatory requirements. The regulatory requirements the staff considered in its review are:

- 10 CFR Part 50, Appendix A, General Design Criteria 10, "Reactor Design" which states:

  "The reactor core and the associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including effects of anticipated operational occurrences."

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR Part 50, Appendix K, which address among others, items such as peak cladding temperature and maximum cladding oxidation.

In its review, the staff also considered the guidance in Standard Review Plan (SRP), Section 4.2, SRP Section II.A(f), which states: "Fuel and burnable poison rod internal gas pressure should remain below nominal system pressure during normal operation unless otherwise justified."

3.0 TECHNICAL EVALUATION

3.1 Fuel Rod Internal Pressure

3.1.1 Background

The CE report CEN-372-P-A provides a basis for a CE-designed plant to change its fuel design criterion for fuel rod internal pressures that states that rod pressures shall not exceed the nominal reactor coolant system pressure during normal or anticipated operational occurrences. The letter from A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, "Acceptance for Referencing CE Topical Report CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," includes the NRC Safety Evaluation Report (SER) for CEN-372. The SER states that licensees seeking to reference CEN-372-P-A should: "1) provide plant-specific LOCA [loss of coolant accident] analyses to determine the impact of maximum calculated rod pressures on cladding rupture timing and peak cladding temperatures, as described in [SER] Section 2.3; and 2) provide analyses for DNB [departure from nucleate boiling] propagation in postulated accidents if the 14x14 steamline break (SLB) is not applicable for maximum rupture strain and percent flow blockage for the licensee's application, as described in [SER] Section 2.4.2." The staff has considered whether the licensee met these criteria for this license amendment request.

In its letter dated December 16, 2002, Entergy proposed a change to the Waterford 3 TSs to add CEN-372-P-A to the list of NRC-approved TRs in TS 6.9.1.11.1. The list in TS 6.9.1.11.1 identified analytical methodologies used in the Waterford 3 Core Operating Limit Report (COLR) to determine cycle-specific operating limits for the parameters that have an important role in maintaining the safety of the plant.

The staff used the acceptance criteria stated in the above referenced letter (April 10, 1990), to review the applicability of CEN-372-P-A to the licensee's rod pressure proposal and the acceptability of the analyses for Waterford 3. The fuel addressed, plant class, and analytical methods used in CEN-372-P-A are all of CE design, as are the Waterford 3 plant, fuel, and analytical methods. The plant characteristics described in CEN-372-P-A envelope those existing at Waterford 3. Therefore the staff concludes, consistent with the SER approving CEN-372-P-A, that Waterford 3 is in the class of plants covered by CEN-372-P-A.

3.1.2 LOCA Analyses

The staff considered whether the licensee’s request addressed plant-specific LOCA analyses to determine the impact of maximum calculated rod pressures on cladding rupture timing and peak cladding temperatures (PCT), and addressed non-LOCA analyses for DNB propagation in

postulated accidents for maximum rupture strain and percent flow blockage. The licensee's submittal addressed both the LOCA and non-LOCA issues, and concluded that the present Waterford 3 analyses continue to bound those which consider the inclusion of the fuel with higher internal pressure. However, the staff requested additional information to verify the licensee's conclusions. The staff raised following issues:

1) To address potential concerns that different generic methodologies might have different analytical sensitivities, the licensee stated that the methods and computer codes (CESEC, CEFLASH, etc.) used in the Waterford 3-specific LOCA analyses producing the limiting results are the same as those used in CEN-372-P-A to produce limiting results in that report. The licensee, in the supplemental letter dated September 29, 2003, clarified that the large break LOCA (LBLOCA) analyses were performed using the LBLOCA methodology described in CENPD-132, Supplement 3-P-A, dated June 1985, and that the small break LOCA (SBLOCA) analyses were performed using the SBLOCA methodology described in CENPD-137, August 1974: Supplement 2-P-A, April 1998 (S2M methodology). The same methodologies are currently referenced in the Waterford 3 TSs and COLR.

2) In a previous letter dated January 21, 2002, the licensee stated that the LOCA methodologies continued to apply specifically by stating that: "Waterford 3 and Westinghouse have ongoing processes that assure that the LOCA analysis input values for peak clad temperature sensitive parameters bound the as-operated plant values for those parameters." This statement assures that the Waterford 3 LOCA methodologies continue to comply with 10 CFR 50.46(c). Therefore, the staff concludes that the LOCA methodologies used to address the LOCA analysis requirements of the CEN-372-P-A SER apply to Waterford 3 for that purpose.

3) The staff requested that the licensee provide the results of the LOCA analyses (PCT, maximum local oxidation, and maximum core-wide hydrogen generation (core-wide oxidation)) for the LBLOCA and SBLOCA analyses it had performed to support the use of higher pressure fuel. In response to the staff's request, the licensee provided the following tables:

EVENT	PCT	Maximum Local Cladding Oxidation	Maximum Core-wide Cladding Oxidation
LBLOCA Analysis using Bounding Fuel Performance Data	2164 ^oF	8.20%	<0.805%
Most Limiting LBLOCA Analysis Final Safety Analysis Report (FSAR) Table 15.6-14)	2177 ^oF	8.55%	<0.805%

EVENT	PCT	Maximum Local Cladding Oxidation	Maximum Core-wide Cladding Oxidation
Most Limiting SBLOCA Analysis (FSAR Table 15.6-14a)	1929 ^{o}F	8.09%	<0.58%
Most Limiting LBLOCA Analysis (FSAR Table 15.6-14)	2177 ^{o}F	8.55%	<0.805%

These analyses demonstrate that the current FSAR licensing basis LBLOCA analysis bounds the LBLOCA analysis considering the internal fuel pressure increased per CEN-372-P-A and SBLOCA analyses. SBLOCA analyses including the increased fuel rod pressure would not be expected to be limiting because of the slower reactor system pressure reduction rate calculated for SBLOCAs.

At the staff's request, the licensee also addressed the concern that the resident fuel may have pre-existing oxidation that needs to be considered in estimating the total LOCA oxidation. In its supplemental letter dated September 29, 2003, the licensee provided a response, including reference to information in CE TR CEN-386-P-A. CEN-386-P-A is an NRC staff-approved methodology for calculating the thickness of pre-existing cladding oxidation. The licensee considered a calculation of the pre-accident cladding oxidation thickness in its assessment of whether the maximum local oxidation criterion is met. The staff concludes from the LOCA analyses results tabulated above, and the information contained in CEN-386-P-A, that the licensee has substantiated its conclusion that the LOCA analyses for Waterford 3 consider the total LOCA oxidation, including pre-existing oxidation, and meet the oxidation criterion of 10 CFR 50.46(b)(2).

The staff also notes that the pre-existing oxidation of the fuel is not expected to contribute to the LOCA maximum core-wide hydrogen generation. Therefore, the staff concludes that the core-wide hydrogen generation analyses results reported above demonstrate that Waterford 3 meets the core-wide hydrogen generation criterion of 10 CFR 50.46(b)(3)

In summary, the licensee has performed LBLOCA and SBLOCA analyses for Waterford 3 using approved Westinghouse/CE methodologies. The licensee's LBLOCA and SBLOCA calculations demonstrate the following:

- The calculated LBLOCA and SBLOCA values for PCT, oxidation, and core-wide hydrogen generation are less than the limits of 2200 ^{o}F, 17%, and 1.0% specified in 10 CFR 50.46(b)(1) through (3), respectively.

- Compliance with 10 CFR 50.46(b)(1) through (3) and (5) assures that the core will remain amenable to cooling as required by 10 CFR 50.46(b)(4). (The staff notes that other matters that could affect coolable geometry are not involved in the requested amendment.)

Therefore, the NRC staff concludes that the licensee's LOCA analyses were performed with LOCA methodologies that apply to Waterford 3 and demonstrate that Waterford 3 complies with the requirements of 10 CFR 50.46(b)(1) through (4). The NRC staff finds the licensee's LOCA

analyses acceptable. Compliance with the long-term cooling requirement of 10 CFR 50.46(b)(5) is not involved in the requested amendment.

3.1.3 Non-LOCA Analyses

The NRC SER for CEN-372-P-A indicates that licensees referencing CEN-372-P-A should provide analyses for DNB propagation in postulated accidents if the 14x14 SLB is not applicable for maximum rupture strain and percent flow blockage for the licensee's application, as described in (SER) Section 2.4.2. CEN-372, Appendix A, provides the results of reactor analyses of main SLB events and locked rotor events for 14x14 and 16x16 fuel designs. These studies demonstrate that the results for 14x14 main SLB events provide bounding results for fuels internally pressurized in accordance with CEN-372-P-A prescriptions. The CEN-372-P-A analyses show that though DNB might occur, the overall effects (e.g., ballooning and time in DNB) would not be sufficient to cause more than only a slight DNB propagation, which would not sustain itself to an unacceptable level.

In its December 16, 2002, submittal, the licensee considered Waterford 3 analyses for four events and compared the results for those analyses to the bounding 14x14 results given in CEN-372-P-A. The four events are:

1) Increased main steam flow with loss-of-alternating current power event,
2) Pre-trip SLB event,
3) Single reactor coolant pump shaft seizure/sheared shaft event, and
4) Control element assembly ejection event.

The Waterford 3 analyses produced times in DNB equal to or less than 8 seconds, which is much less than the times in DNB that the analysis set forth in CEN-372-P-A showed were acceptable. From this, the licensee concludes that the findings in CEN-372-P-A, that DNB would not propagate for non-LOCA events, apply to Waterford 3. In as much as the staff has approved the analyses set forth in CEN-372-P-A, and the licensee has demonstrated that it meets the conditions for applying CEN-372-P-A at Waterford 3, the application is acceptable with respect to DNB propagation.

3.1.4 Summary Regarding CEN-372-P-A

The staff finds that CEN-372-P-A, and its provisions to determine the allowable fuel internal pressure, acceptable for application to Waterford 3, based on confirmatory results of LOCA and non-LOCA analyses that the licensee performed in accordance with the conditions delineated in the staff CEN-372-P-A SER, dated April 10, 1990.

3.2 Implementation of TSTF-363

TSTF-363 is a model for licensees seeking to delete dates, revision numbers, supplement numbers, and amendment numbers for TRs listed in plant TS. Using this model, a licensee would relocate the deleted information to the relevant licensing document; in this case, the plant COLR. The staff has reviewed the licensee's proposal for consistency with the TSTF-363 model and for compliance with applicable requirements of 10 CFR Part 50, specifically, 10 CFR 50.46 for LOCA methodologies.

3.2.1 Implementation of TSTF-363

TSTF-363 models how licensees may cite references to approved TRs in the plant TS using only the report number and title. Under this model, the licensee would include specific identifying information (e.g. report number, title, revision, supplement, amendment, dates, and other designations) in the COLR.

The licensee proposed to implement the TSTF-363 process for the TRs listed below:

1) “The ROCS and DIT Computer Codes for Nuclear Design,” CENPD-266-P-A, April 1983; and “C-E Methodology for Core Designs Containing Gadolinia-Urbana Burnable Absorber,” CENPD-275-P-A, May 1998.
2) “C-E Method for Control Element Assembly Ejection Analysis,” CENPD-0190-A, January 1976.
3) “Modified Statistical Combination of Uncertainties,” CEN-356(V)-P-A, May 1988.
4) “Calculative Methods for the C-E Large Break LOCA Calculation Model for the Analysis of C-E and Westinghouse Designed NSSS [Nuclear Steam Supply System],” CENPD-132, Supplement 3-P-A, June 1985.
5) Calculative Methods for the C-E Small Break LOCA Evaluation Model,” CENPD-137-P, August 1974, Supplement 2-P-A, April 1998.
6) “CESEC - Digital Simulation for a Combustion Engineering Nuclear Steam Supply System,” CENPD-107, December 1981.
7) “Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System,” ENEAD-01-P, Revision 0.
8) “ Fuel Rod Maximum Allowable Gas Pressure,” CEN-372-P-A, June 1988, Supplement 1-P, July 1989.

As a clarification, different report numbers, titles, revision numbers, supplements, amendments, dates, and other designations for LOCA methodologies identify specifically different LOCA methodologies (e.g., CENPD-137, August 1974, Supplement 1-P-A, April 1985, is a different LOCA methodology than CENPD-137, August 1974, Supplement 2-P-A, April 1998, and not merely a different vintage of the same methodology). For LOCA codes, 10 CFR 50.46(a)(1) governs methods of calculating emergency core cooling system performance. The NRC staff reviews each revision, supplement, or amendment to these LOCA code TRs in accordance with Section 50.46. If a new version of such a report is acceptable, the staff generically approves it for use at individual power reactors at which the conditions under which the report would be applied are within the ranges of the conditions for which the report is approved. These approved ranges of conditions, as understood at the time of review, are set forth in the reports and the associated NRC SEs.

With regard to the LOCA methodologies listed above, we find the specific relocations requested above acceptable because relocation of the specific identifying information for these methodologies, which have been approved for Waterford 3 without changing the identifying designations, will preserve the specific identifications of the methodologies and will not introduce unreviewed or unapproved methodologies.

3.2.2 Summary Regarding Implementation of TSTF-363

The staff finds the relocation of the specific identifying TR documentation to the Waterford 3 COLR from the Waterford 3 TS proper and acceptable because the NRC staff approves the referenced reports, the general identifying titles and numbers of these topical reports will be retained in the TS, the specific report-designating information will be preserved in the COLR, and the reports will be applied on a cycle-specific basis within the approved ranges of conditions for their applicability, as set forth above. This is consistent with the TSTF-363 model.

4.0 SUMMARY

Based on the review discussed in Section 3.1, the staff concludes that the reference to CEN-372-P-A and operation of Waterford 3 with fuel pressures determined in accordance with CEN-372-P-A methods, as proposed by the licensee, is acceptable.

Based on the review discussed in Section 3.2, the staff concludes that relocation of specific identifying information (e.g. report number, title, revision, supplement, amendment, dates, and other designations) of the topical reports listed in Section 3.2 from the Waterford 3 TS, to the Waterford 3 COLR references, is acceptable.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 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.

Principal Contributor: F. Orr

Date: October 31, 2003

cc:

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