May 23, 2005

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

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 (WATERFORD 3) - ISSUANCE OF EXIGENT AMENDMENT RE: REMOVAL OF LICENSE CONDITION ON INSTRUMENT UNCERTAINTY (TAC NO. MC6835)

Dear Mr. Venable:

The Commission has issued the enclosed Amendment No. 201 to Facility Operating License No. NPF-38 for Waterford 3. This amendment consists of a change to the operating license in response to your application dated April 27, 2005, as supplemented by letter dated May 12, 2005.

The amendment removes the license condition on instrument uncertainty that was imposed on the Waterford 3 license with the issuance of License Amendment 199 for the extended power uprate.

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/

Thomas W. Alexion, 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. 201 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. 201
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 April 27, 2005, as supplemented by letter dated May 12, 2005, 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, Facility Operating License No. NPF-38 is hereby amended as follows:

   The license condition on instrument uncertainty that was imposed on the operating license with the issuance of License Amendment 199 for the extended power uprate (EPU) on April 15, 2005, is hereby considered complete and is therefore removed. The licensee may now operate the Waterford 3 facility at power levels exceeding 3441 megawatts-thermal (MWt), not to exceed the 3716 MWt that was conditionally authorized with Amendment 199 for the EPU.

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/

David Terao, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: May 23, 2005

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 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 April 27, 2005 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML051190174), Entergy Operations, Inc. (the licensee), requested a change to the operating license for the Waterford Steam Electric Station, Unit 3 (Waterford 3). The supplement dated May 12, 2005 (ADAMS Accession No. ML051370307), 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* on May 5, 2005 (70 FR 23892).

The proposed change would remove the license condition on instrument uncertainty that was imposed on the Waterford 3 license with the issuance of License Amendment 199 for the extended power uprate (EPU).

2.0 REGULATORY EVALUATION

On April 15, 2005, the NRC approved the EPU for Waterford 3. As part of the approval, the NRC imposed the following license condition:

> As stated in the licensee's letter dated February 5, 2005, the licensee committed as follows: "Prior to exceeding 3441 MWt, Entergy will submit, for NRC review and approval, a description of how Entergy accounts for instrument uncertainty for each Technical Specification parameter impacted by the Waterford 3 Extended Power Uprate." Accordingly, subject to completion of this condition, the licensee shall not operate the Waterford 3 facility at a power level exceeding 3441 MWt.

The regulations at Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR) discuss the requirements for technical specifications (TSs) for nuclear power plants. However, no TS changes are proposed with this application. Instead, the application describes how the licensee maintains compliance with 10 CFR 50.36 by accounting for instrument uncertainty for certain TS plant parameters, where appropriate.

3.0 TECHNICAL EVALUATION

While reviewing the Waterford 3 EPU request, the NRC staff noted that the licensee did not always explicitly account for instrument uncertainties when confirming that the required values of TS parameters are satisfied. The NRC staff was concerned that by not explicitly accounting for instrument uncertainties, the possibility exists that TS requirements may not be met in all cases. Therefore, the license condition discussed above was imposed to require the licensee to describe how instrument uncertainties for parameters that are affected by the EPU are accounted for in order to assure compliance with TS requirements, as applicable.

In response, the licensee documented the treatment of instrument measurement uncertainties for parameters that were revised in association with the EPU or were otherwise pertinent to the EPU analyses that were performed (even if the value of the parameter did not change for EPU). The specific parameters that satisfied the following criteria were addressed within the scope of the license condition:

a) The parameter is a value which is measured using plant equipment. That is, the parameter is directly indicated to operators using installed plant instrumentation

and

b) The parameter is a value that is specified by a Limiting Condition for Operation (LCO) of the Waterford 3 TSs. Parameters listed in TS Tables which are called out by LCO's are considered within the scope of this effort. When an LCO refers to values specified in the Core Operating Limits Report (COLR), such values would also be considered within the scope of this effort.

Using the above criteria, the licensee generated a table of pertinent TS parameters and categorized them as follows: Category A is for parameters that have an offset between the TS value and the analytical value that explicitly accounts for instrument uncertainties; Category B is for parameters that have an offset between the TS value and the value that is allowed by surveillance procedures that explicitly accounts for instrument uncertainties; Category C is for parameters that do not have an offset from the analytical value or from the value allowed by the surveillance procedures; and Category D is for parameters that do not have an analytical basis and do not explicitly account for instrument uncertainties. Based on a review of the information that was provided, the NRC staff found that the table appeared to provide a complete listing and assessment of the TS parameters that are included within the scope of the license condition.

The Category A and B parameters explicitly account for instrument uncertainties either in the applicable analyses or in the surveillance procedures, thereby assuring compliance with TS requirements consistent with the staff's expectations. While the Category C parameters do not explicitly account for instrument uncertainties, no TS parameters are included in this category and TS compliance is not an issue. Consequently, the Category D parameters are the only ones that do not explicitly account for instrument uncertainties when confirming that the TS requirements are satisfied. The NRC staff's review of the Category D parameters is as follows.

Per the licensee's submittal there are five Category D parameters (see underlined portions below) in three TSs:

- TS 3.2.3, Azimuthal Power Tilt
  - Action b.2 & b.3, 50% Rated Thermal Power (RTP) and 55% RTP setpoint.
  - Action b.3, 95% RTP.
- TS 3.2.6, Reactor Coolant Cold Leg Temperature
  - Footnote on T_{cold}, allowing an increase to 559 EF for up to 30 minutes following a reactor power cutback.
- TS 3.7.1.7, Atmospheric Dump Valves
  - Action a & b, 70% RTP.

TS 3.2.3, Azimuthal Power Tilt

The power levels listed in TS 3.2.3 are consistent with those in the Standard TSs (NUREG-1432, Standard Technical Specifications, Combustion Engineering Plants). Reducing thermal power to < 50 percent RTP and the trip setpoint to #55 percent RTP provides an acceptable level of protection from increased power peaking due to potential xenon redistribution while maintaining a power level sufficiently high to allow the tilt to be analyzed. Verifying the azimuthal power tilt is within specified limits at a thermal power of $ 95 percent RTP provides an acceptable exit from this action after the measured azimuthal power tilt has been returned to an acceptable value. Additionally, as stated in its EPU submittal (letter dated November 13, 2003; ADAMS Accession No. ML040260321), the licensee explicitly accounts for power measurement uncertainty in its safety analyses. Therefore the NRC staff concludes continued use of these power levels in TS 3.2.3 is acceptable.

TS 3.2.6, Reactor Coolant Cold Leg Temperature

The licensee's TS has a footnote that states, "Following a reactor power cutback in which (1) Regulating Groups 5 and/or 6 are dropped or (2) Regulating Groups 5 and/or 6 are dropped and the remaining Regulating Groups (Groups 1, 2, 3, and 4) are sequentially inserted, the upper limit on Tc [cold leg temperature] may increase to 559EF for up to 30 minutes."

The reactor power cutback system is part of the steam bypass control system (SBCS). The SBCS's purpose is to maximize plant availability by making full utilization of the turbine bypass valves and/or dropping of selected control element assembly (CEA) groups to avoid unnecessary reactor trips. The SBCS responds to load reductions by opening turbine bypass valves. Should the capacity of the turbine bypass valves be exceeded, the SBCS will drop the selected CEA groups to reduce reactor power.

The licensee submitted additional information on May 12, 2005, to support the use of engineering judgement for this parameter. Based on the NRC staff's review, the licensee's engineering judgement is based primarily on the following factors:

> The reactor power cutback system functions in response to a loss-of-load to rapidly reduce reactor power to avoid reactor trips that would otherwise result on high

pressurizer pressure. In this regard the initiation of the reactor cutback system is similar to loss-of-load type events categorized as decreases-in-heat removal by the secondary system in its Final Safety Analysis Report Chapter 15 analysis, and that these events, including reactor power cutbacks, are bounded by the loss-of-condenser vacuum and loss-of-feedwater analyses that were shown to not challenge departure from nucleate boiling ratio (DNBR) margins.

The licensee has performed analyses for control system response to transients involving a reactor power cutback. The example provided in its supplemental letter dated May 12, 2005, is for an end of cycle reactor power cutback to approximately 50 percent power that resulted in a 7 EF rise in T_{cold} without operator action.

The licensee has had this footnote in its TSs since the original license. The 10 EF added to the TS limit on T_{cold} is unchanged. The TS range on T_{cold}, and the footnote temperature have been lowered for EPU.

The Core Protective Calculator (CPC) system initiates automatic protective action to assure that DNBR and local power density fuel design limits are not exceeded. The CPC wide range T_{cold} band extends from 495 EF to 580 EF, encompassing the 559 EF temperature in the footnote.

Since the plant would have already experienced a Chapter 15 event initiator (e.g., loss-of-load or a partial loss-of-feedwater flow) prior to the reactor power cutback, it is not necessary or credible to postulate another event happening during the limited period of time (30 minutes) that the TS 3.2.6 footnote would be applicable after a reactor power cutback.

The NRC staff has reviewed the licensee's documented engineering judgement in its supplemental letter dated May 12, 2005, and has the following additional comments.

The NRC staff considers it reasonable and appropriate to consider another event following a reactor power cutback system activation, especially if there is a potential common mode failure mechanism. As the SBCS is intended to accommodate plant transients without initiating a reactor trip, activation of the SBCS, including reactor power cutback, should not be considered as the initial action in an event. Potential common mode failure mechanisms include, but are not limited to, electrical grid disturbances that cause a load rejection and subsequently lead to a loss-of-offsite power, or the magnitude of a transient associated with a reactor power cutback system activation may result in the loss-of-feedwater control.

The NRC staff considers it reasonable to use the decrease in heat removal by the secondary system analyses to provide insight into the potential impact of having the cold leg temperature being at the 559 EF allowed in the TS 3.2.6 footnote. These analyses indicate there is reasonable assurance of sufficient margin to accommodate the higher T_{cold} when coupled with a reduced reactor power.

The NRC staff considered the total loss of forced reactor coolant flow analysis included in the EPU submittal [licensee letters dated November 13, 2003; May 7, 2004 (ADAMS Accession No. ML041330175); and May 12, 2004 (ADAMS Accession

No. ML041380147)], to provide insight into the potential impact of having the cold leg temperature being at the 559 EF allowed in the TS 3.2.6 footnote. This analysis utilizes full RTP and the minimum cold leg temperature allowed by TS 3.2.6 with uncertainty. The analysis description indicates there are intricacies in the methodology used that make a lower cold leg temperature result in a more adverse DNBR. This analysis indicates there is reasonable assurance of sufficient margin to accommodate the higher T_{cold}.

The licensee has changed its T_{cold} program concurrent with EPU. The new T_{cold} temperature range of 536 EF to 549 EF is reflected in TS 3.2.6. The analytical values include a 3 EF uncertainty to extend the range from 533 EF to 552 EF. The 559 EF value in the TS 3.2.6 footnote is 7 EF above the analytical value. The licensee has stated it expects a reactor power cutback, without operator action, to result in T_{cold} increase of 7 EF. These relationships lend a reasonableness to the nominal 10 EF added to the TS range by their similarity.

The CPC system initiates automatic protective action to assure that DNBR and local power density fuel design limits are not exceeded. This system ensures the minimum initial DNBR is maintained prior to the start of any event.

Based on the licensee's documentation and the NRC staff's review, the NRC staff concludes there is reasonable assurance the T_{cold} footnote in TS 3.2.6 is acceptable.

TS 3.7.1.7, Atmospheric Dump Valves (ADVs)

In the licensee's April 27, 2005, application the 70 percent RTP in TS 3.7.1.7 is listed as a Category D parameter, meaning there is no analytical basis and the number is based on engineering judgement. However, in the licensee's EPU submittals [letters dated November 13, 2003, and July 14, 2004 (ADAMS Accession No. ML042010150)] the small-break loss-of-coolant accident (SBLOCA) analysis takes credit for the ADVs with reactor power greater than 70 percent. TS 3.7.1.7 was specifically added as part of the EPU submittal. In addition, the licensee's EPU submittal specifically says SBLOCA includes a 0.5 percent power measurement uncertainty. In its supplemental letter dated May 12, 2005, the licensee explained why it placed this parameter in Category D. Based on the NRC staff's review, the licensee's rationale is that while the SBLOCA analysis supports the 70 percent reactor power level, other power levels would also show acceptable results and the specific use of 70 percent reactor power as an input to the analysis was based on engineering judgement. Given the description of the SBLOCA analysis in the EPU submittal dated November 13, 2003, and the supplemental information provided by letter dated May 12, 2005, the NRC staff concludes that continued use of 70 percent RTP in TS 3.7.1.7 is acceptable.

In addition, the licensee provided additional discussion regarding four of the Category A TS Sections on moveable CEAs because the explicit offset for uncertainty is applied to CEA worth, which is directly related to CEA position indication. The NRC staff's review of this additional discussion is as follows.

The four Category A parameters on moveable CEAs are combined into two sections in the licensee's April 27, 2005, application that address CEA position uncertainty. These sections are similar with respect to CEA position uncertainty:

- Section 4.1, CEA Misalignment Criteria (19"), TS 3.1.3.1 ACTIONs b, c, and d.

- Section 4.2, CEA Insertion Limits (145" and COLR [Core Operating Limits Report] Figures 4 and 5), TS 3.1.3.1 ACTION f, TS 3.1.3.5, and TS 3.1.3.6.

In its submittal the licensee described how CEA position uncertainty is inherently captured in the computational CEA worth uncertainties used in its current methodology. Computational CEA worth uncertainties are determined by comparisons between computed and measured CEA worth. CEA position uncertainty is one factor contributing to the difference between measured and computed CEA worth. The computational CEA worth uncertainties are defined to bound a 95/95 tolerance limit (95 percent probability with 95 percent confidence) about the population of total difference between the computed and measured CEA worth. The computational CEA worth uncertainties are then applied conservatively to ensure the most adverse effect of CEA worth during analysis.

The NRC staff has reviewed this approach and compared it with similar activities. The NRC staff previously reviewed and approved the use of the CASMO-4/SIMULATE-3 computer codes at Arizona Public Service Company's (APS's) Palo Verde Nuclear Generating Station, Units 1, 2, and 3 (NRC letter dated March 20, 2001; ADAMS Accession No. ML010860187). The description of CEA position uncertainty capture above is consistent with the description of CEA position uncertainty capture in APS's submittal to allow the use of CASMO-4/SIMULATE-3 computer codes (APS letter dated June 8, 2000; ADAMS Accession Nos. ML003723799 and ML010440094). The NRC staff has determined that continuing to capture CEA position uncertainty inherently in the computational CEA worth uncertainties is a reasonable approach and consistent with industry practice. Therefore, the NRC staff concludes capture of CEA position uncertainty inherently in the computational CEA worth uncertainties is acceptable.

Evaluation Summary

As discussed above, the NRC staff finds that the licensee's treatment of instrument uncertainties is consistent with the NRC staff's expectations for assuring compliance with TS requirements. Accordingly, the NRC staff finds that the licensee's request to remove the license condition for addressing instrument uncertainty that was imposed on the Waterford 3 license with the issuance of License Amendment 199 for the EPU on April 15, 2005, should be approved.

4.0 EXIGENT CIRCUMSTANCES

The amendment request was submitted on an exigent basis because the need for a license amendment to remove the license condition was not recognized by Entergy or the NRC staff until just prior to the issuance of the EPU on April 15, 2005. Following notification, the licensee worked expeditiously to provide the NRC staff with the April 27, 2005, follow-up license amendment request. The licensee requests approval of the proposed amendment by May 27, 2005, to support power ascension and avoid derating of Waterford 3 following restart from the spring 2005 refueling outage. Therefore, the licensee requested that this proposed license amendment be considered under exigent circumstances as described in 10 CFR 50.91(a)(6).

Based on the above circumstances, the NRC finds that the licensee used its best efforts to make a timely application as soon as it was informed that a license amendment request would be needed to remove the license condition, and could not have avoided the need for the exigency. The NRC also finds that, in light of these circumstances, the licensee and the Commission must act quickly and time does not permit the Commission to publish a *Federal Register* notice allowing 30 days for prior public comment. As set forth below, the NRC has determined that this amendment involves no significant hazards consideration. Based on the foregoing, the NRC finds that exigent circumstances exist as defined in 10 CFR 50.91(a)(6), with regard to the license amendment requested by the licensee's application dated April 27, 2005.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue. The NRC staff's analysis is set forth below.

The proposed change does not result in a change to any structure, system, or component (SSC) and no new accident precursors are created. Therefore, the proposed change does not involve a significant increase in the probability of previously evaluated accidents. The proposed change has no impact on the safety analysis because the application of an explicit offset to the TS parameters for instrument uncertainty provides additional assurance that the plant will operate within the operating envelope previously analyzed. The completion of the license condition will allow Waterford 3 to operate at the power level of 3716 megawatts-thermal (MWt) that has previously been evaluated and approved by the NRC staff as documented in Amendment 199 to the Waterford 3 Operating License. Therefore, the accident mitigation features of the plant for previously evaluated accidents are not affected by the proposed change. Accordingly, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not change the design function or operation of any SSC. The proposed change introduces no new mode of operation nor involve any new plant equipment. The proposed change does not affect the functional capability of safety-related equipment. The completion of the license condition will allow Waterford 3 to operate at the power level of 3716 MWt that has previously been evaluated and approved by the NRC staff as documented in Amendment 199 to the Waterford 3 Operating License. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change does not result in a change to any SSC. The accident mitigation features of the plant for previously evaluated accidents are not affected by the proposed change. The proposed change has no impact on the safety analysis because the application of an explicit offset to the TS parameters for instrument uncertainty provides additional assurance that the plant will operate within the operating envelope previously analyzed. Existing TS operability and surveillance requirements are not reduced by the proposed change. The completion of the license condition will allow Waterford 3 to operate at the power level of 3716 MWt that has previously been evaluated and approved by the NRC staff as documented in Amendment 199 to the Waterford 3 Operating License. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above considerations, the NRC staff concludes that the amendment meets the three criteria of 10 CFR 50.92(c). Therefore, the NRC staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

6.0 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.

7.0 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 changes 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 May 5, 2005 (70 FR 23892). 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.

8.0 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.

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Date: May 23, 2005

Waterford Steam Electric Station, Unit 3

cc:

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May 2005