UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 26, 2015

Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – REQUEST FOR ADDITIONAL INFORMATION REGARDING THE INCREASED CONTROL ELEMENT ASSEMBLY DROP TIMES LICENSE AMENDMENT REQUEST (TAC NO. MF6459)

Dear Sir or Madam:

By letter dated July 2, 2015, as supplemented by letter dated August 14, 2015, Entergy Operations, Inc. proposed changes to the Waterford Steam Electric Station, Unit 3, Technical Specification 3.1.3.4, "CEA [Control Element Assembly] Drop Time," and Final Safety Analysis Report, Chapter 15, "Accident Analyses," due to an increase in the average control element assembly drop times discovered during required testing.

After reviewing your request, the U.S. Nuclear Regulatory Commission staff has determined that additional information is required to complete the review. Please provide the additional information requested in the enclosure within 30 days of the date of this letter.

If you have any questions, please contact me at 301-415-3229 or michael.orenak@nrc.gov.

Sincerely,

Michael D. Orenak, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure:
Request for Additional Information

cc w/enclosure: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST REGARDING THE

CONTROL ELEMENT ASSEMBLY DROP TIMES

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

By letter dated July 2, 2015, as supplemented by letter dated August 14, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML15197A106 and ML15226A346, respectively), Entergy Operations, Inc. (the licensee) submitted a license amendment request (LAR) in which it requested a revision to Technical Specification (TS) 3.1.3.4, "CEA [Control Element Assembly] Drop Times," and the Final Safety Analysis Report (FSAR) for the Waterford Steam Electric Station, Unit 3. The proposed change would modify TS 3.1.3.4 and the FSAR, Chapter 15 accident analyses to account for a 0.2 second increase in the CEA drop times. Questions 1 through 20 were developed as a result of the U.S. Nuclear Regulatory Commission (NRC) staff's review of the July 2, 2015, LAR. Question 21 was developed as a result of the NRC staff's review of the August 14, 2015, supplement.

1. In Attachment 2, page 4, the following statement is made:

> The analysis margin for the axial power distribution was reduced from an ASI [axial shape index] of +0.3 to +0.2, which is conservative to the COLR [core operating limits report] limit of +0.16.

Table 15.0-4 of the FSAR shows the axial shape index used is: $-0.2 \leq ASI \leq +0.2$. Provide the bases for the current margin for the ASI of +0.3 and explain why this is different from the FSAR value.

2. In Attachment 2, pages 4, 6, 16, and 17, the following statement is made:

> The analysis margin for the least negative Doppler reactivity was reduced from -0.00113 to -0.0013 $\Delta\rho/\sqrt{°K}$.

Provide the bases for the least negative Doppler coefficients of -0.00113 to -0.0013 $\Delta\rho/\sqrt{°K}$. Also, justify adequacy of the use of the value of -0.0013 $\Delta\rho/\sqrt{°K}$ in the reanalysis used to support the proposed increase in the limits of the CEA drop times in TS 3.1.3.4.

Enclosure

3. In Attachment 2, page 7, the following statement is made:

> Plant changes since the extended power uprate have been incorporated into the analysis under the 10 CFR [Title 10 of the *Code of Federal Regulations*] 50.59 process. The analysis has been updated to account for the replacement steam generators (SGs) and the NGF [next generation fuel] DNBR [departure from nucleate boiling ratio] correlation. The revised evaluation started with the analysis of record and only revised CEA drop time to determine the impact.

Clarify the difference between "analysis" and "analysis of record." Does the revised evaluation include the replacement SGs and NGF DNBR correlation?

4. In Attachment 4, page 4, the Figure 2, "Cycle 15 through Cycle 20 CEA Insertion Times," shows the repeated measured CEA insertion times for Cycle 20. The measured insertion times for Cycle 20-2 are significantly shorter than that of Cycle 20-1.

Discuss the measurement methods for the Cycle 20-1 and Cycle 20-2 data, including uncertainties, identifying the causes for the reduction of the insertion times from measurements for Cycle 20-1 to Cycle 20-2, and justify the adequacy of the Cycle 20-2 data by showing that the reduced insertion times are not contributed from random measurement errors. Was anything done to improve the drop time between Cycle 20-1 and Cycle 20-2 or was the test simply repeated?

5. Table 4.0-1 in Attachment 1, page 4, provides the new CEA drop times. Other than the change at 90 percent CEA insertion (from 3.0 to 3.2 seconds), what is the basis for the other changes to the CEA drop time curve? Was any plant data used to determine the new curve? If so, provide plant data and demonstrate that the new curve is conservative. Provide the reactivity versus time (or CEA position) curve that was presented during the pre-application public meeting on April 22, 2015.

6. Attachment 1, page 12, calls out References 7.40, 7.41, and 7.42. However, these three references are not provided in the References section (pages 13-15). It appears that some of the references are improperly numbered. For example, page 12 of Attachment 1 states, "The NRC approved this request in NRC Technical Specification Amendment 158 [Reference 7.42]." However, this appears to be Reference 7.37 on page 15. Update the reference numbers in text so they are consistent with the reference list.

7. Attachment 1, pages 2 and 3, discusses apparent and potential causes for increased rod drop times. Provide the cycles as to when plant modifications were made (i.e., replacement SGs were installed between Cycles XX and YY).

8. Attachment 1, page 6, paragraph 3, states:

> In order to aid the NRC review, the relevant Waterford 3 licensing basis history is provided. The last major plant change that submitted the

> accident and transient analyses to the NRC for review was the extended power uprate [Reference 7.15] and alternate source term implementation [Reference 7.18]. Two additional major plant changes that have been implemented since the extended power uprate are the use of Next Generation Fuel (NGF) and Steam Generator replacement. These changes were addressed by NRC approvals (Table 4.0-2) and the 10 CFR 50.59 process. The Westinghouse reload analysis methodology has been applied to implement these Waterford 3 changes.

Please confirm that the stated "Westinghouse reload analysis methodology" was previously approved by the NRC and that no changes were made to the approved methodology since it was used to implement the use of NGF and SG replacement. If changes were made to the NRC-approved reload analysis methodology, identify and justify the changes for the use in support of the proposed TS regarding the CEA drop times.

9. Attachment 2, page 5, FSAR, Section 15.1.2.3, "Increased Main Steam Flow," states:

> The increase in peak secondary pressure is based on the loss of condenser vacuum results which showed an increase in peak secondary pressure of less than 1 psi.

FSAR, Section 15.1.2.3 does not discuss a loss of condenser vacuum (LOCV). Explain why a LOCV was used for this case.

10. Attachment 2, page 6, states:

> The increase in peak primary pressure was based on the loss of condenser vacuum results which showed an increase in peak primary pressure of less than 1 psi [pounds per square inch] when the pressurizer pressure exceeded the pressurizer safety valve opening setpoints (2575 psia [pounds per square inch absolute]). The increase in peak secondary pressure was based on the loss of condenser vacuum results which showed an increase in peak secondary pressure of less than 1 psi.

FSAR, Section 15.1.2.4 does not discuss LOCV. Explain why LOCV was used for this case.

11. Attachment 2, pages 4 and 5, FSAR, Section 15.1.2.3 considers two cases: (1) typical case and (2) worst departure from nucleate boiling performance case. This section does not discuss which case was analyzed. Clarify if both cases were considered in the updated analysis.

12. The table in Attachment 2, page 6, contains an "*" after 2584 psia. Provide the significance of the "*".

13. Attachment 2, page 26, states, in part, that changes to the updated analysis include:

> ...the use of the actual initial thermal margin reserved in the LCO.

Provide the parameter that was changed and the original and updated values.

14. In FSAR, Section 15.9.1.1, "Asymmetric Steam Generator Transient," there are four events considered, including: Loss of Load to One Steam Generator (LL/1SG), Excess Load to One Steam Generator (EL/1SG), Loss of Feedwater to One Steam Generator (LF/1SG), and Excess Feedwater to One Steam Generator (EF/1SG). In the updated analysis described in Attachment 2, pages 25-26, verify if all four cases were analyzed or if it was assumed the limiting case was still limiting after the rod drop time change.

15. In Attachment 2, the results of the analysis for a loss of normal feedwater (LONF) event were used to support the adequacy of the dose releases for the following events:

- Page 10 – Steam System Piping Failures: Pre-Trip Power Excursion Analysis
- Page 18 – Single Reactor Coolant Pump (RCP) Shaft Seizure/Sheared Shaft
- Page 22 – CEA Ejection
- Page 23 – Primary Sample or Instrument Line Break
- Page 24 – SG Tube Rupture
- Page 24 – Loss of Coolant Accident

Provide justification for each event above for the use of the LONF results to support the radiological releases.

16. Attachment 2, page 17, discussed the analysis of the single reactor coolant pump (RCP) shaft seizure/sheared shaft events. The reanalysis showed that the peak primary pressure was increased by 20 pounds psi to 2442 psia, and the peak secondary pressure was increased by 1 psi to 1118 psia. The increase in the peak primary pressure was based on the LONF long-term results and the increase in the peak secondary pressure was based on the LOCV results.

Please provide justification for the use of two different events results, LONF and LOCV, to derive the peak primary and secondary pressure, respectively, for the RCP shaft seizure/sheared shaft events.

17. The fourth paragraph in Attachment 2, page 7, states, in part:

> The steam generator blowdown is not impacted by the revised CEA drop time...

The SG blowdown rate was dependent on the SG pressure; however, the impact of the revised CEA drop time on the SG pressure response was not sufficiently discussed in the LAR. Provide a more detailed discussion on the impact of the revised CEA drop time on the SG pressure response.

18. The second to last paragraph in Attachment 2, page 9, states:

> The loss of feedwater flow event demonstrated that the impact of the revised CEA drop time on long term parameters is insignificant. Hence there is an insignificant impact on the plant cooldown to shutdown cooling conditions post-trip.

Provide justification for the use of the results of the LONF flow event to the steam line break long-term cooldown.

19. The fourth paragraph in Attachment 2, page 16, states, in part:

> The increase in peak primary pressure was based on the loss of feedwater long term results which showed an increase in peak primary pressure of less than 20 psi...

Provide justification for the use of the LONF long-term results to the total loss of forced reactor coolant flow event in determining the peak primary pressure.

20. The second to last paragraph in Attachment 2, page 18, states, in part:

> The loss of condenser vacuum event peak primary and secondary pressure increases would bound that expected for the uncontrolled CEA withdrawal from subcritical conditions.

Further, the second to last paragraph in Attachment 2, page 19, states, in part:

> The loss of condenser vacuum event peak primary and secondary pressure increases would bound that expected for the uncontrolled CEA withdrawal from low power condition.

Provide justification for the use of the LOCV results to both cases of the CEA withdrawal events in determining the peak primary and secondary pressure increases due to the CEA drop time changes.

21. Attachment 1, page 2, of the August 14, 2015, supplement provides the CEA drop time curve for the 0.8 second holding coil decay time (Curve 3). This curve has the initial CEA motion delayed 0.2 seconds over the revised curve (Curve 2). However, by 1.15 seconds, both curves are identical. Given that the CEAs insert via gravity, justify how the CEAs that start dropping at 0.8 seconds (Curve 3) are at the same position as CEAs that started dropping at 0.6 seconds (Curve 2) by 1.15 seconds (10 percent inserted).

August 26, 2015

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Sincerely,
/RA/
Michael D. Orenak, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
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ADAMS Accession No.: ML15232A275

OFFICE	NRR/DORL/LPL4-2/PM	NRR/DORL/LPL4-2/LA	NRR/DSS/SRXB/BC
NAME	MOrenak	PBlechman (LRonewicz for)	CJackson
DATE	8/25/15	8/20/15	8/25/15
OFFICE	NRR/DORL/LPL4-2/BC	NRR/DORL/LPL4-2/PM	
NAME	MKhanna	MOrenak	
DATE	8/25/15	8/26/15	

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