February 7, 2008

Mr. Kevin T. Walsh
Vice President of Operations
Entergy Operations, Inc.
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - REQUEST FOR ADDITIONAL INFORMATION RE: SUPPLEMENT TO THE EMERGENCY CORE COOLING SYSTEM PERFORMANCE ANALYSIS SUBMITTAL IN SUPPORT OF NEXT GENERATION FUEL (TAC NO. MD6954)

Dear Mr. Walsh:

By letter dated August 9, 2007, as supplemented on October 4, 2007, Entergy Operations, Inc. (Entergy, the licensee), submitted information on the application of the optional spacer grid steam cooling heat transfer model in the Westinghouse ECCS [Emergency Core Cooling System] Performance Evaluation Model for CE [Combustion Engineering] Plants (1999 EM) Large Break Loss-of-Coolant Accident (LBLOCA) ECCS Performance Analysis for the implementation of CE 16x16 Next Generation Fuel Assemblies in Waterford Steam Electric Station, Unit 3.

After reviewing your request, the U.S. Nuclear Regulatory Commission staff has determined that additional information is required to complete the review. This was discussed with your staff and it was agreed that Entergy would provide the additional information requested in the enclosure within 30 days from the receipt of the formal request for additional information.

If you have any questions, please contact me at (301) 415-1480.

Sincerely,

/RA/

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

Docket No. 50-382

Enclosure: Request for Additional Information

cc w/encl: See next page

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

SUPPLEMENT TO THE ECCS [EMERGENCY CORE COOLING SYSTEM]

PERFORMANCE ANALYSIS SUBMITTAL

IN SUPPORT OF NEXT GENERATION FUEL

WATERFORD STEAM ELECTRIC STATION, UNIT 3

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

Entergy Operations, Inc. (Entergy), the licensee for Waterford Steam Electric Station, Unit 3, by its letters dated August 9, 2007, and October 4, 2007, provided submittals on Emergency Core Cooling System [ECCS] Performance Analysis, and Supplement to the ECCS Performance Analysis in Support of Next Generation Fuel and requested the review of the same.

To support the U.S. Nuclear Regulatory Commission (NRC) assessment of the acceptability of the submittals, please provide the additional information outlined below.

Reference Letter: October 4, 2007, "Supplement to the ECCS Performance Analysis Submittal in Support of Next Generation Fuel in Waterford 3 – 1999 EM Optional Steam Cooling Model Justification."

1. Figure 3.3-7 does not show quench. Please show the results of the calculation through quench for Figure 3.3-7 and also the peak local oxidation plot until quench in Figure 3.3-2.

2. Please explain why once the spacer grids quench, the steam flow does not return to the lower steam flow consistent with the steam flow without grids dashed line in Figure 3.3-4.

3. How does this model compare with FLECHT reflood data (in particular, steam flow and heat transfer coefficient) for the tests when the reflood rate is near 1 inch per second and the pressure is lowest?

4. Do axial regions above the rupture node to the top of the core remain at intermediate temperatures (that are lower than the peak cladding temperature (PCT)) sufficiently long to cause the oxidation to also approach 17 percent? Please provide the clad temperatures above the rupture node out to quench along with the oxidation percentage.

5. Is thermal radiation to the grids modeled from the surrounding rods above and below the grid spacers? If not please explain why it is neglected.

6. Does Figure 3.3-4 include flow redistribution after rupture? Please explain.

7. Is there any heat transferred to the entrained drops above the quench front? How is this heat transfer modeled? Please explain.

8. Is Baker-Just applied at all temperatures? Does the Baker-Just reaction rate constant capture the data over the full range of temperatures down to and including quench? Please explain.

Section 4.0 of the supplementary large break LOCA analysis in the October 4, 2007, letter stated that the analysis of record (AOR) results included in the August 9, 2007, letter are unchanged by the supplementary LBLOCA results using the NRC-approved version of the optional steam cooling model. In support of the above, please address the following issues:

9. Table 3.1.1 listed the key results of a large break loss-of-coolant accident (LBLOCA) analysis for the limiting local oxidation case using the NRC-approved optional steam cooling model. Please list the results similar to those in Table 3.1.1 for the same limiting case analyzed by the earlier version of the optional steam cooling model, and demonstrate that the AOR results calculated by the earlier version of the model are not or slightly different from the results in Table 3.1.1.

10. The reanalyzed LBLOCA cases, the local cladding oxidation case (on page 3 of the supplementary LOCA analysis) and PCT case (on page 14), are the limiting cases identified in the AOR analysis that was performed with the earlier version of the optional steam cooling model. Provide the results of an analysis to demonstrate that those two cases remain to be the limiting cases considering postulated LBLOCA cases of different sizes, locations, initial conditions and single failure assumptions when the NRC-approved version of model is used to perform LBLOCA analyses.