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Ken Peters
Director, Nuclear Safety Assurance
Waterford 3

W3F1-2003-0056

July 30, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Supplement to Amendment Request
Addition of CEN-372 to Technical Specification 6.9.1.11.1
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES: 1. Entergy letter dated December 16, 2002, *Use of CEN-372-P-A* (W3F1-2002-0104)

Dear Sir or Madam:

In Reference 1, Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specifications (TSs) to add the topical report entitled "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A, to the list of analytical methods in TS 6.9.1.11.1 used to determine the Waterford 3 core operating limits.

In a conference call on June 10, 2003, Entergy and members of your staff discussed four questions developed during the technical review of this request. The Entergy response to these questions is provided in the attachment to this letter.

There are no technical changes to the original amendment request proposed here. The no significant hazards consideration discussion included in Reference 1 is not affected by any information contained in this supplemental letter. There are no new commitments contained in this letter.

If you have any questions or require additional information, please contact Jerry Burford at 601-368-5755.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 30, 2003.

Sincerely,

KJP/FGB/ssf

Attachment:
Response to Request For Additional Information

cc: Thomas P. Gwynn, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office
Louisiana DEQ/Surveillance Division
American Nuclear Insurers

Attachment 1

To

W3F1-2003-0056

Response to Request for Additional Information

Response to Request for Additional Information

Question 1

The letter provides a maximum Peak Clad Temperature (PCT) value (2164 °F) for the limiting large break loss of coolant accident (LOCA) analysis, but it does not provide maximum calculated cladding oxidation and hydrogen generation values for this analysis (or for the corresponding limiting large break LOCA events, if different than the limiting PCT large break LOCA). Please provide this information or justify not providing it.

Response 1

The PCT value of 2164 °F resulted from a large break LOCA analysis that used bounding fuel performance data as input, including data where fuel rod internal gas pressure exceeded the nominal Reactor Coolant System pressure of 2250 psia. The maximum cladding oxidation determined from this analysis was 8.20%. The maximum core-wide cladding oxidation was found to be less than 0.805%. Final Safety Analysis Report (FSAR) Table 15.6-14 provides the results for the most limiting LOCA analysis for Waterford. The data for the two analyses is provided in the table below:

Event	Peak Clad Temperature	Max Cladding Oxidation	Max Core-Wide Cladding Oxidation
Large Break LOCA Analysis using bounding fuel performance data	2164 °F	8.20%	< 0.805%
Most Limiting Large Break LOCA Analysis (FSAR Table 15.6-14)	2177 °F	8.55%	< 0.805%

Question 2

The letter does not provide limiting PCT, oxidation, and hydrogen generation values for the limiting small break LOCA events. Please provide this information or justify that the large break LOCA analysis is bounding for each of these items.

Response 2

The results of small break LOCA analyses are reported in FSAR Table 15.6-14a. The limiting small break LOCA analysis is bounded by the limiting large break LOCA analysis reported in FSAR Table 15.6-14. The data for both events is provided in the table below:

Event	Peak Clad Temperature	Max Cladding Oxidation	Max Core-Wide Cladding Oxidation
Most limiting small break LOCA Analysis (FSAR Table 15.6-14a)	1929 °F	8.09%	< 0.58%
Most limiting large break LOCA Analysis (FSAR Table 15.6-14)	2177 °F	8.55%	< 0.805%

Question 3

Why was pre-trip SLB chosen for Waterford 3? Was feed line break considered? If not, why not? If it was, why is it not limiting? The staff requests that you justify why the SLB is limiting for Waterford 3 and discuss why a feed line break analysis would not be limiting.

Response 3

The maximum cladding strain and maximum percent flow blockage occurs for the pre-trip Steam Line Break (SLB) as presented in Appendix A of CEN-372-P-A. This is the most limiting event for the purpose of Departure from Nucleate Boiling (DNB) propagation (longest time in DNB with the greatest cladding strain and flow blockage). The postulated accidents for Waterford 3 that result in DNB were reviewed. Since the time in DNB conditions for each of these events (including pre-trip SLB) is significantly less than the time in DNB for the pre-trip SLB in CEN-372-P-A, the Waterford 3 analyses are bounded by that analysis. The Feedwater Line Break event was not considered because it does not result in DNB and, therefore, is not a concern for DNB propagation.

Question 4

The methodologies used to analyze LOCAs and non-LOCA events for Waterford 3 are different than those used to support CEN 372-P-A. The staff requests that you justify that the same events are limiting for Waterford 3 as for CEN 372-P-A.

Response 4

The Waterford 3 analyses presented in support of this amendment use the same methods and computer codes (CESEC, CEFLASH, etc.) as were used in CEN 372-P-A. In the future, methods may be changed and the limiting event for time in DNB may be different. However, the postulated accidents for Waterford 3 will continue to be evaluated using approved methods that conservatively predict the onset of DNB. Key parameter values (e.g., time in DNB) for all Waterford 3 accidents will continue to be compared to those used in the conservative cladding strain calculations in CEN-372-P-A to demonstrate that DNB propagation does not occur.