

NRC Section Chief: Robert A. Gramm.
Entergy Operations Inc., Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request:
 December 16, 2002.

Description of amendment request:
 The proposed amendment will add the topical report entitled "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A, to the list of analytical methods in Technical Specification (TS) 6.9.1.11.1 used to determine the Waterford Steam Electric Station, Unit 3 (Waterford 3) core operating limits.

Basis for proposed no significant hazards consideration determination:
 As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: No.

The proposed change does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident. The proposed change adds an NRC [Nuclear Regulatory Commission]-approved topical report to the list of analytical methods used to determine the core operating limits. The effect of the addition of this new reference is to revise the fuel design criterion for internal rod pressure to accept rod pressures that may exceed nominal Reactor Coolant System operating pressure. The use of this revised criterion continues to ensure that the consequences of an accident remain within acceptable limits. The change also proposes the administrative deletion of report date and revision levels in the list of references. These changes do not alter any of the assumptions or bounding conditions currently in the Final Safety Analysis Report.

Waterford 3 performed a large break loss-of-coolant accident (LOCA) analysis using bounding fuel performance data as described in CEN-372-P-A. This analysis concluded that the peak cladding temperature remained within 10 CFR 50.46 limits.

In addition to the LOCA analysis, an evaluation of the potential for departure from nucleate boiling (DNB) propagation was performed as described in CEN-372-P-A. The results confirmed that Waterford 3 is bounded by the results evaluated in the topical report and that DNB propagation will not occur.

Based on these analyses, there is no increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not involve any change to the configuration or method of

operation of any plant equipment that is used to mitigate the consequences of an accident. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting failure been identified as a result of the proposed change. The intent of the proposed change is to reference an NRC-approved topical report in the Technical Specifications. The topical report justifies an acceptance criterion that allows fuel rod internal pressure to exceed RCS [reactor coolant system] pressure. There are no new accidents created by this change. An administrative aspect of this change, the deletion of date and revision levels, was also considered and does not create a new or different accident.

The impact of fuel rod internal pressure exceeding reactor coolant system (RCS) pressure was considered in both an emergency core cooling system (ECCS) performance analysis and in a DNB propagation evaluation performed for Waterford 3. These two aspects were required considerations based on the NRC Safety Evaluation review of the topical report. The results demonstrated that Waterford 3 continues to meet 10 CFR 50.46 and that there is no potential for DNB propagation.

Based on these analyses, there is no possibility of the creation of a new or different kind of accident from those previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change adds an NRC-approved topical report to the list of analytical methods used to determine core operating limits. It also deletes the revision number and dates associated with each of the topical reports listed. The effect of the addition of the new reference is to revise the fuel design criterion for fuel rod internal pressure to accept rod pressures that may exceed nominal RCS operating pressure. The use of this revised criterion continues to ensure that the consequences of an accident remain within acceptable limits. Since the core operating limits will continue to be established by an NRC-approved methodology and the results will be verified to meet the established acceptance criteria of 10 CFR 50.46, the change will provide adequate core protection. Thus, the proposed amendment does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: N. S. Reynolds, Esquire, Winston & Strawn 1400 L Street, NW., Washington, DC 20005-3502.

NRC Section Chief: Robert A. Gramm.

Entergy Operations Inc., Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request:
 December 20, 2002.

Description of amendment request:
 The proposed amendment makes several administrative changes to the Waterford Steam Electric Station, Unit 3, Technical Specifications (TSs) to revise, delete, correct, or clarify certain titles, page numbers, and heading information. The proposed amendment also revises personnel and committee titles that have been changed, revises administrative reporting requirements to conform to 10 CFR 50.4, and deletes redundant or unnecessary requirements from TSs 5.4, 6.6, and 6.7.

Basis for proposed no significant hazards consideration determination:
 As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes are primarily to correct titles, page numbering errors, and otherwise make the TS index pages consistent with other NRC [U. S. Nuclear Regulatory Commission] approved pages. These changes are all of an administrative nature and have no effect on any plant equipment or structures. Therefore, these changes do not increase the probability or consequences of an accident previously evaluated.

The proposed amendment also deletes TS 5.4.1 and 5.4.2. Values for RCS [Reactor Coolant System] design pressure, temperature, and volume are contained in the Final Safety Analysis Report. Any changes to these are controlled by 10 CFR 50.59. Therefore, removing the section from the TS will not increase the probability or consequences of previously evaluated accidents.

The proposed amendment also deletes TS 6.6 and 6.7, and revises TS 6.9.1 and TS 6.9.2 to administratively conform reporting requirements to those in 10 CFR [part] 50. Therefore, removing these sections from the TS will not increase the probability or consequences of previously evaluated accidents.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes are administrative in nature and do not involve a physical alteration of the plant. No new or different equipment or modes of operation are being introduced by this proposed change. Thus, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.