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Waterford 3

W3F1-2002-0104

December 16, 2002

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

SUBJECT: Waterford 3 Steam Electric Station, Unit 3
Docket 50-382
License Amendment Request NPF-38-245
Use of CEN-372-P-A

Dear Sir or Madam·

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests an amendment for Waterford Steam Electric Station, Unit 3 (Waterford 3) to revise the Technical Specifications (TS). This proposed amendment is 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. This topical report was previously approved in an Nuclear Regulatory Commission Safety Evaluation dated April 10, 1990. While the general approach was found acceptable, the NRC noted that licensees would need to submit supporting analysis in order to use it at their facility. In addition, the deletion of report dates and revision numbers for all of the reports listed is also requested This latter change is consistent with approved Technical Specification Task Force (TSTF) traveler TSTF-363.

The proposed amendment has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards consideration. The basis for this determination is included in the attached submittal.

Please note that page 6-20a is also being changed by another License Amendment Request. When replacement pages are requested, Entergy will provide a version of Technical Specification Page 6-20a that reflects the latest approved version.

The proposed change does not involve any new commitments Entergy requests approval of the proposed amendment by October 1, 2003 so that it can be implemented for Waterford 3's Fall 2003 refueling outage. Once approved, the amendment will be implemented within 60 days.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 16, 2002.

Sincerely,

J. E. Venable
Vice President, Operations
Waterford Steam Electric Station, Unit 3

JEV/FGB/cbh

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)

cc: Mr. Ellis W. Merschoff, NRC Regin IV
Mr. N. Kalyanam
NRC Resident Inspectors Office
J. Smith (Wise Carter)
N. S. Reynolds
Louisiana DEQ / Surveillance Section
American Nuclear Insurers

Attachment 1

W3F1-2002-0104

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford 3).

The proposed change will revise the Technical Specifications (TS) to modify the list of analytical methods used to determine the core operating limits to include a new reference to a topical report that justifies operation with higher fuel rod internal pressures. In addition, this change will also delete the identification of dates and revision numbers for the topical reports identified in this section. This latter change is consistent with approved Technical Specification Task Force (TSTF) traveler TSTF-363.

2.0 PROPOSED CHANGES

The proposed change affects Waterford 3 TS 6.9.1.11.1 and requests:

- The addition of a topical report reference to CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure." The new entry will also include a reference to the technical specification which reflects the use of this report; in this case, it is TS 3.2.1, Linear Heat Rate.
- The deletion of applicable dates or revisions for the other topical reports listed as items 1 through 7 of this specification. This change is consistent with approved TSTF-363.

The proposed amendment to Section 6.9.1.11.1 would add a new topical report to the list of analytical methods used to determine the core operating limits. This topical report justifies a revised fuel design criterion regarding fuel rod internal pressures. The new reference is to Topical Report CEN-372-P-A, which has been previously approved by the NRC staff (as documented in a letter dated April 10, 1990, from Ashok C. Thadani, Director, Division of Systems Technology, Office of Nuclear Reactor Regulation, to A.E. Scherer, Combustion Engineering (CE), "Safety Evaluation of Combustion Engineering Topical Report CEN-372-P, Fuel Rod Maximum Allowable Gas Pressure") as an acceptable basis for a new fuel rod internal pressure criterion.

3.0 BACKGROUND

Technical Specification Section 6.9.1.11.1 lists the analytical methods previously reviewed and approved by the NRC that are used to determine the core operating limits. Plant operation is limited in accordance with the values of cycle specific parameter limits that are established in the Core Operating Limits Report (COLR) using these NRC approved analytical methods. The COLR is a licensee-controlled document that establishes the appropriate cycle-specific core limits for each cycle.

The NRC includes a fuel design acceptance criterion for rod pressures in the Standard Review Plan (SRP), Section 4.2. SRP Section II.A.1(f) states: "Fuel and burnable poison rod internal gas pressures should remain below nominal system pressure during normal operation unless otherwise justified." This had been implemented by the various fuel designers (e.g., see CE Topical Report CENPD-269-P, which included a fuel design criterion that rod internal pressure would not exceed nominal reactor coolant system (RCS) pressure during normal or anticipated operational occurrences (AOOs)). C-E revised the rod internal pressure criterion to permit rod pressures to exceed nominal RCS pressure. The revised criterion is described in CE Topical Report CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure." The report also presents analyses for demonstrating that the new design criterion is met. This revised criterion is summarized as: The fuel rod internal hot gas pressure shall not exceed the critical maximum pressure determined to cause an outward clad creep rate that is in excess of the fuel radial growth rate anywhere locally along the entire active fuel length of the fuel rod.

The analysis approach proposed by CE for maintaining a stable fuel cladding gap is to establish a critical rod pressure limit for their fuel designs that is conservative. Fuel performance analyses confirm that the maximum rod internal pressures in any given fuel batch will not exceed this proposed critical rod pressure limit. The rod pressure analysis methodology used to determine rod pressures is the same as that used in CENPD-269-P when the critical pressure limit was set equal to the RCS pressure.

Increased internal fuel rod pressure is a result of many factors including burnup time for fuel assemblies, utilization of low leakage cores, mechanical modifications to fuel pellets and rod configuration, and the use of Erbia as a burnable absorber. The NRC, in their Safety Evaluation (SE) of the CE report CEN-372-P dated April 10, 1990, approved the use of this report by licensees. However, licensees referencing the topical report are required to:

1) Provide plant specific loss-of-coolant accident (LOCA) analysis to assure that Peak Cladding Temperature (PCT) continues to remain below the limits required by 10 CFR 50.46 for these rods. Westinghouse committed to reduce either the LOCA linear heat rate (LHR) or the critical rod pressure limit (defined within the Topical Report) if necessary to assure that the limits were met.

2) Provide analysis for departure from nucleate boiling (DNB) propagation in postulated accidents if the bounding 14 x 14 steam line break is not applicable for calculating maximum cladding rupture strain and percent flow blockage.

The change to reference the topical report is requested to utilize the revised fuel internal pressure criterion.

4.0 TECHNICAL ANALYSIS

This technical analysis discussion will address both aspects of the NRC requirements for use of this topical report. This will include a discussion of the emergency core cooling system (ECCS) performance analysis and of the DNB propagation potential.

ECCS Performance

ECCS performance evaluations are performed for each cycle using the NRC approved models as currently described in the references in Technical Specification Section 6.9.1.11.1.4 and 6.9.1.11.1.5.

In anticipation of rod internal pressure exceeding RCS pressure, a large break LOCA analysis for Waterford 3 has been performed using bounding fuel performance data. This bounding data included fuel rod internal pressures that exceeded RCS pressure that was generated by the FATES3B computer code consistent with Topical Report CEN-372-P-A. The ECCS performance analysis explicitly calculated the Peak Cladding Temperature (PCT) over the range of burnups that the hot rod could operate at the Peak Linear Heat Generation Rate. The maximum PCT was 2164 °F, which is less than the PCT reported in the FSAR (2177 °F) and below the limits given in 10 CFR 50.46.

The Waterford 3 cycle-specific reload analyses, utilizing methods and limits specified in the COLR, ensure that the PCT continues to remain below the limits required by 10 CFR 50.46 (see TS 6.9.1.11.2).

<u>DNB Propagation</u>

The maximum cladding strain and percent flow blockage of the bounding 14 x 14 pre-trip steam line break (SLB) presented in Appendix A to CEN-372-P-A is the basis for the evaluation of the potential for DNB propagation for Waterford 3. The limiting transient presented in Appendix A to CEN-372-P-A established a limiting time in DNB (see page A-13 of the topical report). For both 14 x 14 and 16 x 16 fuel, as long as the time in DNB is less than the limiting pre-trip SLB case reported in CEN-372-P-A, then DNB propagation is not a concern.

A review of postulated accidents for Waterford 3 determined that the bounding 14 x 14 SLB bounds the maximum cladding rupture strain and percent flow blockage. The postulated accidents allowing fuel in DNB were reviewed to determine the length of time that DNB may occur. These postulated accidents are:

1. Increased main steam flow with loss-of-alternating current power (LOAC) event
2. Pre-trip SLB event
3. Single reactor coolant pump shaft seizure/sheared shaft event
4. Control Element Assembly (CEA) ejection event

The length of time in DNB for these events are presented below:

- The time for fuel to remain in DNB for the increased main steam flow with LOAC event is less than 8 seconds due to a reactor trip generated by the Core Protection Calculator (CPC).

- The time in DNB for the limiting pre-trip SLB event is less than 5 seconds due to a reactor trip generated by the CPC.

- The time in DNB for the limiting sheared shaft case is less than 5 seconds. The time in DNB for the single reactor coolant pump shaft seizure event is also less than 5 seconds because of the immediate trip on CPC pump speed.

- The time in DNB for the CEA ejection event is much less than 8 seconds due to a trip very early in the event and to a strong Doppler feedback component. Even for a bounding time in DNB as great as 8 seconds, DNB propagation would not occur for this event.

Since the length of time in DNB for all of the events evaluated for DNB propagation is substantially less than the limiting value of CEN-372-P-A, DNB propagation is not a concern for Waterford 3.

Deletion of the Revisions / Dates

In a letter from the NRC to Siemens Power Corporation dated December 15, 1999, the NRC agreed that it would be acceptable to cite references to the approved topical reports using report number and title information only in the Technical Specifications. This was intended to permit licensees to utilize the latest NRC-approved version of a topical report without having to require a change to the Technical Specifications every time a report was revised. This has the effect of eliminating burden and unnecessary expenditure of both NRC and licensee resources when preparing for new cycle-specific analysis and licensing. As a part of the acceptance of this approach, it was noted that the Core Operating Limits Report (COLR) itself would provide the complete identification of each of the references used to prepare the COLR.

This letter was used as the basis for a generic change initiated by the Boiling Water Reactor Owners Group (BWROG) that evolved into TSTF-363. This generic change was determined to be applicable to the improved standard specifications for each of the Owners Groups and was approved by the NRC. This TSTF was incorporated into Revision 2 of NUREG-1432, *Standard Technical Specifications for Combustion Engineering Plants,* for example.

Entergy finds that this change does not impact safe operation and is considered to be an administrative change. Waterford 3 COLR currently documents the complete identification information (i.e., including revision or date) for each of the topical reports used in its preparation.

Conclusion

The Waterford 3 analyses provided herein fulfill the NRC Safety Evaluation Report (SER) requirements for referencing Topical Report CEN-372-P-A. DNB propagation will not occur as a result of fuel rod fission gas pressure exceeding RCS pressure for the Waterford 3 16 x 16 fuel design. The LOCA analysis results remain below the limits of 10 CFR 50.46. Therefore, Entergy requests NRC approval to include a reference to the approved Topical Report CEN-372-P-A in the Waterford 3 Technical Specifications. In addition, the change to delete topical report revision information from the Technical Specifications has no impact on safety and is considered to be an administrative change. It is consistent with the current improved Standard Technical Specifications.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. The changes are consistent with previously NRC approved positions: the revised criterion of the topical report was accepted in a Safety Evaluation dated April 10, 1990, and the deletion of the revisions and dates was accepted via the generic technical specification change process. The use of the new criterion continues to meet applicable regulations, including 10 CFR 50, Appendix A, General Design Criterion (GDC) 10, Reactor Design, and 10 CFR 50.46.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any GDC differently than described in the FSAR.

5.2 No Significant Hazards Consideration

Entergy Operations, Inc. (Entergy) proposes to amend the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specifications (TS) to revise the list of reports used in preparing the Core Operating Limits Report (COLR). The changes include:

- The addition of a topical report reference to "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A. This report was approved by the NRC in a Safety Evaluation dated April 10, 1990. The new entry will also include a reference to TS 3.2.1, Linear Heat Rate, which reflects the use of this report.
- The deletion of applicable dates and revision levels for the other topical reports listed as items 1 through 7 of this specification. This change is considered to be an administrative change and is consistent with approved Technical Specification Task Force (TSTF) traveler TSTF-363.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an 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. The proposed change adds an NRC-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 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.

Based on the above, Entergy concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

The change to add a reference to Topical Report CEN-372-P-A has been reviewed and approved for other facilities, including:

Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – request approved in Amendments 101, 89, and 72, respectively, dated October 4, 1995.

San Onofre Nuclear Generating Station, Units 2 and 3 – request approved in letter dated August 3, 1990. Note that this approval was for plant-specific use of the topical and did not involve a revision to any Technical Specification pages.

Attachment 2

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Proposed Technical Specification Changes (mark-up)

ADMINISTRATIVE CONTROLS

INDUSTRIAL SURVEY OF TOXIC OR HAZARDOUS CHEMICALS REPORT

6.9.1.9 Surveys and analyses of major industries in the vicinity of Waterford 3 which could have significant inventories of toxic chemicals onsite to determine impact on safety shall be performed and submitted to the Commission at least once every 4 years.

6.9.1.10 A survey of major pipelines ($\geq$ 4 inches) within a 2-mile radius of Waterford 3, which contain explosive or flammable materials and may represent a hazard to Waterford 3, including scaled engineering drawings or maps which indicate the pipeline locations, shall be performed and submitted to the Commission at least once every 4 years.

CORE OPERATING LIMITS REPORT COLR

6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or any remaining part of a reload cycle.

6.9.1.11.1 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as follows:

~~1) "The~~ ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, ~~April 1983~~; and "C-E Methodology ~~for Core~~ Designs Containing Gadolinia-Urania Burnable Absorber," CENPD-275-P-A, ~~May 1988.~~ (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, ~~3.1.1.3 for~~ MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

2) "C-E Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, ~~January 1976.~~ (Methodology for Specification 3.1.3.6 for Regulating and group P CEA Insertion Limits and 3.2.3 for Azimuthal Power Tilt).

3) "Modified Statistical Combination of Uncertainties" CEN-356(V)-P-A, ~~May 1988.~~ (Methodology for Specification 3.2.4 for DNBR Margin and 3.2.7 for ASI).

4) "Calculative Methods for the C-E Large Break LOCA Calculation Model For The Analysis of C-E and W Designed NSSS," CENPD-132, Supplement 3-P-A, ~~June 1985.~~ (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt and 3.2.7 for ASI).

5) "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," CENPD-137-P, ~~August 1974.~~ Supplement 2-P-A, ~~April 1998.~~ (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt and 3.2.7 for ASI).

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT COLR (Continued)

6) "CESEC - Digital Simulation for a Combustion Engineering Nuclear Steam Supply System," CENPD-107~~, December 1981,~~ (Methodology for Specification 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating and group P CEA Insertion Limits, and 3.2.3 for Azimuthal Power Tilt).

7) "Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System," ENEAD-01-~~P, Revision 0,~~ (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, ~~3.1.1.3 for MTC~~, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

8) "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A. (Methodology for Specification 3.2.1, Linear Heat Rate).

6.9.1.11.2 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.11.3 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 Not Used