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Waterford 3

W3F1-93-0099
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PR

December 14, 1993

Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Technical Specification Change Request NPF-38-148

Gentlemen:

The attached description and safety analysis supports changes to the Waterford 3 Technical Specifications that will:

1. remove the reactor vessel material specimen withdrawal schedule from the Waterford 3 Technical Specifications, and

2. update the Waterford 3 Reactor Coolant System pressure-temperature (P-T) curves.

The first proposed change (above) is being submitted in an effort to eliminate unnecessary duplication of update submittals associated with the Waterford 3 reactor vessel material specimen withdrawal schedule. Currently, since the withdrawal schedule is part of the Technical Specifications, updates to the schedule have to be submitted as a license amendment request (before implementation). Additionally, Section II.B.3 of Appendix H to 10 CFR Part 50 also requires the submittal of a proposed change to the schedule prior to implementation.

The proposed change to remove the withdrawal schedule from the Technical Specifications is consistent with guidance provided in NRC Generic Letter 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel

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Material Specimens from Technical Specifications." The schedule is currently included in the Waterford 3 Final Safety Analysis Report (FSAR) Table 5.3-10. The proposed changes to the specimen pull schedule, will be incorporated in the FSAR.

The second proposed change (P-T Curve update) to the Waterford 3 Technical Specifications is being submitted as a result of Combustion Engineering's reanalysis utilizing inputs from the testing results of the Reactor Vessel specimen material pulled April 11, 1991. A copy of the P-T Curve and Reactor Vessel specimen withdrawal schedule analysis report is provided as Attachment C of this submittal.

In addition to the current Technical Specification change request, Reactor Vessel material specimen withdrawal schedule updates are being submitted to the NRC under separate cover letter in accordance with Section II.B.3 of Appendix H to 10 CFR Part 50.

The proposed changes described herein have been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that the proposed changes involve no significant hazards considerations. The Plant Operating Review and Safety Review Committees have reviewed and accepted these proposed changes based on the foregoing evaluation.

Should you have any questions or comments, please contact Paul Caropino at (504) 739-6692.

Very truly yours,

RPBarkhurst

R.P. Barkhurst
Vice President, Operations
Waterford 3

RPB/PLC/dc
Attachments: Affidavit
NPF-38-148

cc: J.L. Milhoan, NRC Region IV
D.L. Wigginton, NRC-NRR
R.B. McGehee
N.S. Reynolds
NRC Resident Inspectors Office
Administrator Radiation Protection Division
(State of Louisiana)
American Nuclear Insurers

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the matter of	)	
	)	
Entergy Operations, Incorporated	)	Docket No. 50-382
Waterford 3 Steam Electric Station	)	

AFFIDAVIT

R.P. Barkhurst, being duly sworn, hereby deposes and says that he is Vice President Operations - Waterford 3 of Entergy Operations, Incorporated; that he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached Technical Specification Change Request NPF-38-148; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

R.P. Barkhurst

R.P. Barkhurst
Vice President Operations - Waterford 3

STATE OF LOUISIANA	)	
	)	ss
PARISH OF ST. CHARLES	)	

Subscribed and sworn to before me, a Notary Public in and for the Parish and State above named this 14TH day of DECEMBER, 1993.

Steven E. Farkas

Notary Public

My Commission expires WITH LIFE.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-38-148

This proposed change modifies the following Waterford 3 Technical Specifications (TS) as described below.

1. TS 3.4.1.3 the double asterisks footnote is revised as follows:

   **A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to ~~285~~ 272 °F unless...

2. TS 3.4.1.4 the double asterisks footnote is revised as follows:

   **A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to ~~285~~ 272 °F unless...

3. Surveillance Requirement 4.4.8.1.2 is revised as follows:

   4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the Reactor Vessel material surveillance program - withdrawal schedule in ~~Table 4.4-5~~ FSAR Table 5.3-10...

4. Surveillance Requirement 4.4.8.1.2 (a & b) is revised as follows:

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the Reactor Vessel material surveillance program - withdrawal schedule in ~~Table 4.4-5~~ FSAR Table 5.3-10. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3. ~~The adjusted reference temperature resulting from neutron irradiation shall be calculated based on the greater of the following:~~

- ~~a. Actual shift in the RT_{NDT} as measured by impact testing of 88114/0145 weld metal;~~
- ~~b. Predicted shift in RT_{NDT} for E8018/BOCA weld metal as determined by Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."~~

5. Figure 3.4-2 is replaced.

6. Figure 3.4-3 is replaced.

7. TS Table 4.4-5 is deleted and replaced with a blank page stating "Table 4.4-5 This Table has been deleted".

8. TS 3.4.8.3 is revised as follows:

   Applicability: Mode 4 when the temperature of any RCS cold leg is less than or equal to ~~285~~ 272 °F#, MODE 5, and...

9. Bases, page B 3/4 4-7 paragraph 2 will be revised as follows:

   The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4 4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper and nickel content of the material in question, can be predicted using FSAR Table 5.3-1 and the recommendations of Regulatory Guide 1.99, Revision 2 ~~1~~, ~~"Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."~~ "Radiation Embrittlement of Reactor Vessel Materials" The heatup and cooldown limit curves...

10. Bases, page B 3/4 4-7 paragraph 3 will be revised as follows:

    The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82 and 10 CFR Part 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in FSAR Table ~~4.4-5~~ 5.3-10...

11. Bases, page B 3/4 4-10 paragraph 3 will be revised as follows:

    The OPERABILITY of the shutdown cooling system relief valve or an RCS vent opening of greater than 5.6 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to ~~285~~ 272 °F...

12. Bases, page B 3/4 4-10 paragraph 4 will be revised as follows:

The restrictions on starting a reactor coolant pump in MODE 4 and with the reactor coolant loops filled in MODE 5, with one or more RCS cold legs less than or equal to ~~285~~ 272 °F, are provided...

13.) Bases, page B 3/4 4-10 paragraph 5 will be revised as follows:

~~The automatic isolation setpoint of the shutdown cooling isolation valves is sufficiently high to preclude inadvertent isolation of the shutdown cooling relief valves during a pressure transient.~~

Existing Specification

See Attachment A

Proposed Specification

See Attachment B

Background

[For Items 1.), 2.), 6.), 11.) and 12.) above]

Selection of 272 °F as the LTOP alignment temperature is consistent with the original value (285 °F) selected. The Appendix G curve (P-T Curve) that ABB/CE generated (during analysis of reactor vessel specimen test results) is less restrictive than the original curve. This is due to improved analysis techniques, including linear elastic fracture mechanics. The original curve crossed the 2500 PSIA pressure line at 285 °F, while the proposed curve crosses at 272 °F. Requiring the alignment of the LTOP system at this temperature ensures that pressure relief protection of the P-T curve is available at all modes of operation. This is consistent with the intent of Branch Technical Position RSB 5-2. Although 272 °F provides additional margin for the plant, it is a result of removing unnecessary conservatism from the P-T Curves.

[For Items 3.), 7.) and 10.) above]

The NRC issued Generic Letter 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications", dated January 4, 1991. This generic letter informed licensees of a case (Joseph M. Farley Nuclear Plant) wherein the NRC approved

a request to remove the subject schedule from the Technical Specifications for that plant. The NRC classified this as line-item improvement consistent with the Commission Policy Statement on Technical Specification improvements. The purpose for the allowed change was to eliminate an unnecessary duplication of requirements. In addition to the Technical Specification requirements for reporting upgrades to the subject schedule, Section II.B of Appendix H to 10 CFR Part 50 requires the submittal to, and approval by, NRC of a proposed withdrawal schedule before implementation. Entergy is requesting NRC approval for adopting the aforementioned line-item Technical Specification improvement.

[For Item 4.)]

Supplement 1 of the staffs' Safety Evaluation Report (SER NUREG-0787) subsection 5.3.1.2 dated October 1981, discusses an exemption to the 10CFR50 Appendix H (II.b) requirements that required Waterford 3 to evaluate weld metal by use of actual impact tests of the weld metal 88114/045 or the predicted shift of weld metal E 880818/BOLA per Regulatory Guide 1.99. These requirements were incorporated into Technical Specification as 4.4.8.1.2 items a and b. This specific exemption was no longer required as a result of the 1983 changes to Appendix H. This fact appears in supplement 8 of NUREG-0787 subsection 5.3.1.2. Therefore, this proposed change removes TS 4.4.8.1.2a and b as an administrative change.

[For Items 5.) and 6.) above]

The first Waterford 3 Reactor Vessel material test specimen capsule was pulled during Refuel 4 (April 11, 1991). The specimen capsule was sent to Babcox and Wilcox Nuclear Systems (B&WNS) for testing. The resulting report stated that the current Pressure-Temperature curves may be extended to 10.5 EFPY based on the fluence calculations performed for capsule W-97 and the Charpy impact test results. It was further reported that based on current operations, Waterford 3 should reach 8 EFPY during cycle 7 in early 1995. Current Technical Specification Figures 3.4-2 and 3.4-3 require that the curves be updated prior to 8 EFPY of operations. Test results were submitted to the NRC on November 25, 1992 (W3F1-92-0369). Note that the NRC had granted an extension beyond the one year requirement for submitting test results. In Entergy Letter W3F1-92-0369, a commitment was made to submit proposed changes to the curves and schedules to the NRC as a Technical Specification amendment request by December 31, 1993.

[For Item 9.)]

Waterford 3 informed the NRC in Letter W3P88-1980 that the plant had applied the methodology of Regulatory Guide 1.99, Rev. 2. This was in response to

Generic Letter GL 88-11, that encouraged Licensees to use the methods described in Revision 2 of Regulatory Guide 1.99 to predict the effect of neutron radiation on reactor vessel materials. This proposed Technical Specification change reflects the above commitment.

[For Item 13.)]

Deletion of wording associated with the automatic isolation setpoint of the shutdown cooling isolation valves is justified since the Shutdown Cooling Auto Closure Interlock feature was removed in 1991 via DC-3260. The proposed Technical Specification change is consistent with the November 1988 revision of Branch Technical Position RSB 5-2 "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures." Branch Position 10 indicates that if pressure relief is from a low pressure system, not normally connected to the primary system, the over pressure protection function should not be defeated by interlocks which would isolate the pressure system from the primary coolant system. (See BTP ICSB3). The Waterford 3 design change (DC-3260) was completed in response to Generic Letter 88-17. The change was formally submitted to the NRC in letter W3P90-0234 dated July 25, 1990, associated with Technical Specification 4.5.2.d.1 change request. The current proposed deletion should have been included with that earlier Technical Specification change.

Description

The requested Technical Specification change at Waterford 3 involves the deletion of Table 4.4-5 and deleting the reference to the table number in a surveillance requirement (Surveillance 4.4.8.1.2) to avoid duplication of submittal requirements. Future updates to the Waterford 3 Reactor Vessel material surveillance program withdrawal schedule will continue to be submitted to the NRC for approval prior to implementation in accordance with the Section II.B.3 of Appendix H to 10 CFR 50 requirement. The withdrawal schedule is currently part of the Waterford 3 FSAR (Table 5.3-10) and will be updated.

The requested Technical Specification change also submits proposed changes to the Reactor Coolant System Pressure/Temperature curves. A report describing development of the requested curve updates is provided as Attachment D.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Although the Reactor Vessel material specimens withdrawal schedule will be removed from the Technical Specifications, the Technical Specifications bases will continue to provide background information on the use of the data obtained from material specimens. Also, updates to the schedule will continue to be submitted to the NRC for approval prior to implementation.

   Operating the plant in accordance with the new, updated P-T Curves will assure preserving the structural integrity of the reactor vessel over the life of the plant. The pressure and temperature limits were developed in accordance with 10 CFR 50 Appendix G requirements.

   Removing the requirements associated with the previous exemption to Appendix H (TS 4.4.8.1.2 items a & b) is purely an administrative change.

   Therefore, the proposed changes will not significantly increase the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

   Response: No.

   Removal of the Reactor Vessel material specimen schedule from the Technical Specifications has no impact on accidents at the plant. Updates to the schedule will still be required to be submitted to the NRC prior to implementation per Section II.B.3 of Appendix H to 10 CFR Part 50.

   Also, updates to the P-T Curves will not create a new or different type accident. The reactor vessel beltline P-T limits were revised applying the general guidance of the ASME Code, Appendix G procedures with the

necessary margins of safety for heatup, cooldown and inservice hydro test conditions.

The change to TS 4.4.8.1.2 items a & b is purely administrative.

Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: No.

   Removal of the schedule for Reactor Vessel material specimen withdrawal from the Technical Specifications does not impact the margin of safety. The schedule will continue to receive NRC review and approval prior to implementation of updates to the schedule.

   Updates to the P-T Curves are provided to preserve the margin to safety to assure that when stressed under operating, maintenance, and testing the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

   The change to TS 4.4.8.1.2 items a & b is purely administrative.

   Therefore, the proposed changes will not result in a significant reduction in the margin of safety.

Safety and Significant Hazards Determination

Based on the above safety analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC final environmental statement.