CENPD-288-NP-RAI
(NRC TAC No. M89663)

ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel: Response to Request for Additional Information

ABB Combustion Engineering Nuclear Operations

ABB

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1 INTRODUCTION

This supplemental report contains responses to the NRC Requests for Additional Information regarding Reference B1 which was transmitted to ABB by the NRC letter identified in Reference B2. They will be included in the approved version for Reference B1 as Appendix B.

2 QUESTIONS AND RESPONSES

NRC Question B1

It should be noted that the allowable stress intensities given on pg. 11 are for elastic analysis; what allowables are used if the component is analyzed plastically?

ABB Response to Question B1

ABB BWR fuel always has been shown to meet design requirements using elastic analysis, which is acceptable under Section F-1322 of Appendix F (Reference B3). In the event that an elastic-plastic analysis were required to demonstrate compliance with the design bases for a specific application, the analysis would be submitted to the NRC under separate application.

NRC Question B2

Why is fatigue evaluation not addressed for the component for the combination of seismic and LOCA loads?

ABB Response to Question B2

CENPD-288-P (Reference B1) provides the methodology for evaluating BWR fuel subjected to postulated Safe Shutdown Earthquake (SSE) and Loss of Coolant Accident (LOCA) events. Therefore Level D Service Limits apply and a fatigue evaluation is not required. [Proprietary Information]

In addition, a fatigue evaluation is performed for normal operating loads as required by Reference B4. This evaluation is discussed in CENPD-287-P (Reference B5).

NRC Question B3

On pg. 12, a statement is made that buckling of the channels is precluded when the maximum calculated stresses do not exceed the allowable stress intensities. A separate buckling evaluation, such as is prescribed in F-1331.5 of Appendix F of the ASME Code Section III, is needed to assure that buckling is precluded. Is a separate buckling evaluation done?

ABB Response to Question B3

Testing of the channel is performed to demonstrate that the channel will not buckle when subjected to seismic plus LOCA loads and deflections. The seismic plus LOCA loads do not cause significant compressive (axial) loads on the fuel channel. The potential collapse mode of concern is local plate buckling caused by excessive lateral deflection.

The channel acceptance criteria given in Section 4.2.2 of Reference B1 is clarified here. It is restated:

"The channel deformation design acceptance criteria are:

(1) The maximum stresses calculated analytically do not exceed the allowable stress intensities.

(2) The maximum deflection of the channel will be compared to values for which safe insertion of control rods can be demonstrated.

(3) The maximum stresses and deflections calculated do not exceed limits for channel weld strength and channel buckling based on test loads and deflections representative of a seismic plus LOCA event."

The channel weld strength and channel buckling limits are typically measured in the test noted in Section 6.1.2.2 of Reference B1 (page 49). [Proprietary Information]

<u>NRC Question B4</u>

The stress limits listed on pg. 11 infer that the analysis performed is elastic. If stresses in the channel structure exceed the yield stress, then the deflections would not be calculated accurately. An accurate determination of the deflection would in this case require plastic analysis. An accurate calculation for the deflection would be needed to assure safe insertion of the control rods. Is it assured?

<u>ABB Response to Question B4</u>

In the methodology described in CENPD-288-P (Reference B1), an elastic analysis is performed, which assures safe insertion of the control rods. It is expected that the calculated primary stresses in the channel will remain elastic so that the calculated deflections are accurate. [Proprietary Information]

<u>NRC Question B5</u>

On pg. 18 It is stated that if the fuel assembly lifts it will then impact against the core support plate. It is stated that the resulting fuel vertical load would range from 2 to 5 g. It is not explained how this loading was determined. If the fuel assembly and the core support plate are very stiff structures, it is conceivable that the impact loading could be significantly higher than 2 to 5 g. This would, of *course, depend on the energy of the fuel assembly at impact. How was the impact deceleration determined?*

ABB Response to Question B5

The range of peak vertical loads with fuel assembly lift given in Section 5.1.1.4 of Reference B1, are from a census of several plant licensing base analyses as reported in the plant specific Final Safety Analysis Report. The FSAR reported peak vertical loads were determined by the plant original equipment supplier for the operating utility. For example, Reference B6 reports a fuel assembly peak vertical fuel assembly load of 4.9 g, using the methodology of Reference B7.

[Proprietary Information]

NRC Question B6

It is stated on pg. 27 that fast neutron irradiation has little effect on the yield and ultimate strengths of the stainless steel and increases the yield and ultimate strengths of the Zircaloy. Hence, it is concluded that the use of unirradiated properties is conservative. This conclusion, however, does not account for possible effects of irradiation on the material ductility. The ASME stress limits given on pg. 11 allow the material to be stressed beyond yield on the basis that the material is ductile. To use these stress limits, it must be assured that the material maintains this ductility when irradiated. Has it been established that the material will maintain ductility at the irradiation levels experienced?

ABB Response to Question B6

The methodology described in CENPD-288-P (Reference B1) will ensure that irradiated fuel assemblies will not fracture if subjected to SSE plus LOCA loads. For example, a SVEA-96 fuel assembly is composed of a Zircaloy-4 channel, Zircaloy-2 fuel rods, and stainless steel castings for the bundle and channel end pieces. The stainless steel castings are of relatively solid design and have very large strength margins. Therefore, the focus of this response concerns the Zircaloy components.

Calculated fuel assembly stresses are compared to stress limits based on minimum material properties for unirradiated material. This is conservative because irradiation of Zircaloy significantly increases its yield stress and ultimate strength. The yield and tensile strengths are expected to increase by factors of at least 2 and 1.5 , respectively, at relatively low irradiation values. See for example, Figure A-18 of Reference B8. Since these irradiated yield stress values exceed the acceptance stress intensity values given in Table 6.1C of Reference

B1, the calculated primary stresses will remain below the irradiated yield stress and ensure margin against fracture. Hence, the reduction in material ductility with irradiation is not of concern.

NRC Question B7

It is stated on pg. 27 that the test load applied to determine stress for the spacer grid and channel weld was cyclic force. Is the number of load cycles applied in the test representative of the number of cycles expected in the actual installation?

ABB Response to Question B7

Yes, the number of load cycles applied in the test bounds the number of cycles expected at the actual plant. The seismic spacer grid and channel weld test is performed with a greater number of load cycles than what is expected during an SSE plus LOCA event. [Proprietary Information]

NRC Question B8

On pg. 29, has the fuel rod performance code VIK been approved by the NRC?

ABB Response to Question B8

The VIK-II code is described in CENPD-285-P (Reference B8). It is the same computer code that is used for the applications described in CENPD-288-P (Reference B1). The Licensing Topical Report CENPD-285-P is under review by the U.S. NRC and is expected to be approved in 1995.

NRC Question B9

On pg. 29, the methods used to combine stresses from different loadings on the fuel rods are described. Since deformation on the channel is important, a description of the method used to determine the total deflection should also be provided. This should include the methods used if the structure goes plastic. What is the method used to determine the total deflection?

ABB Response to Question B9

The method used to calculate the total deflection of the channel is described in Section 5.1.4.1 (page 28) of CENPD-288-P (Reference B1). [Proprietary Information]

NRC Question B10

On pg 15, sect. 5.1.1.1, Discussion: Does the model of fig. 5.2 include the confined fluid?

ABB Response to Question B10

Yes, the confined fluid is included in the model shown in Figure 5.2. The model is discussed in more detail in Section 6.3.1.1 of CENPD-288-P (Reference B1). [Proprietary Information]

NRC Question B11

On pg 31, sect, 5.2.1, Methodology, steps (1), (2) & (3):

(A) How is bounding arrived at in step (1)?

(B) What is the significance of not going through step (2) in all cases, irrespective of the results from step (1)?

(C) More information is necessary to establish the validity of step (3), Please provide.

ABB Response to Question B11

Section 5.2.1 of Reference B1 provides the methodology for demonstrating that ABB BWR reload fuel assemblies will remain engaged in the lower support structure following a postulated SSE/LOCA event. [Proprietary Information]

ABB Response to Question B11, Item (A)

[Proprietary Information]

ABB Response to Question B11, Item (B)

[Proprietary Information]

ABB Response to Question B11, Item (C)

[Proprietary Information]

NRC Question B12

On pg. 31, sect. 5.2.1, Discussion, step (3):

A technical justification for step (3) is required.

ABB Response to Question B12

[Proprietary Information]

NRC Question B13

On pg. 58, sect. 6.3.2.1, GOBLIN/DRAGON Model:

Are the results of the three channel and one channel representations significantly different? How do they compare?

ABB Response to Question B13

[Proprietary Information]

NRC Question B14

On pg. 69, sect. [SVEA-96 Fuel Assembly Vertical Acceleration]:

*The significance of the last * paragraph is not evident. Clarification is required.*

ABB Response to Question B14

This question refers to the fuel assembly lift calculation performed for Example 2, in Section 6.4 of Reference B1. [Proprietary Information]

[Proprietary Information]

NRC Question B15

On pg. 77, sect. 6.5.1.2, Resident Fuel Assembly Seismic response, (also applies to sects. 6.5.1.3 & 6.5.2):

Further explanation is required to establish the validity of the conclusion(s).

ABB Response to Question B15

Section 6.5 of Reference B1 discusses the methodology to be used when the information concerning seismic and LOCA excitation for a specific plant is limited. Additional explanation of this methodology is provided below in response to this question.

[Proprietary Information]

* The original NRC question had a typographical error "elastic" should be "last".

NRC Question B16

The integrity evaluation of a mixed core is not addressed. Is there a possibility of mixed core being used? If so, how is it integrity established?

ABB Response to Question B16

Most operating reactors have a core composed of a mix of fuel designs, either by the same vendor or from several vendors. The seismic/LOCA evaluation methodology described in Reference B1 considers this fact when performing an evaluation for a specific fuel design in a particular plant.

[Proprietary Information]

3 REFERENCES

B1. ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel, CENPD-288-P, NRC transmittal letter NFBWR-94-018, May 26, 1994

B2. NRC Facsimile Transmission from S. L. Wu (NRC) to D. Ebeling-Koning (ABB), June 15, 1995

B3. ASME Boiler and Pressure Vessel Code, Section III, Appendix F, 1992 Edition

B4. ASME Boiler and Pressure Vessel Code, Section III, Part NB, 1992 Edition

B5. Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, ABB Report CENPD-287-P (proprietary), CENPD-287-NP (nonproprietary), June, 1994.

B6. River Bend Station Updated Safety Analysis Report, Section 3.9, Table 3.9B-2aa, August 1987.

B7. BWR Fuel Assembly Evaluation of Combined Safe Shutdown (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (Amendment No. 3), GE Report NEDO-21175-3-A, October 1984.

B8. Fuel Rod Design Methods for Boiling Water Reactors, ABB Report CENPD-285-P (proprietary) CENPD-285-NP (non-proprietary), May 1994.

B9. ASME Boiler and Pressure Vessel Code, Section III, Appendix II-1520, 1992 Edition

B10. U.S. NRC Standard Review Plan Section 4.2, Appendix A, NUREG-0800, July 1981.

B11. Y. Sasaki, Y. Sasaki, H. Niwa, "Dynamic Analysis of Fuel Elements in Boiling Water Reactor," Vol. D, Paper 4-7, 4th Inter. Conf. on Structural Mechanics in Reactor Technology, 1977.

B12. T. Ikeda, et al., "Analysis of Response of BWR Core Structures to Earthquakes," Journal of Nuclear Science and Technology, August 1984.

B13. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification, ABB Report RPB-90-93-P-A (proprietary), RPB-90-91-NP-A (nonproprietary), October 1991.

B14. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity, ABB Report RPB-90-94-P-A (proprietary), RPB-90-92-NP-A (nonproprietary), October 1991.

B15. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96, CENPD-283-P (proprietary), CENPD-283-NP (nonproprietary), March 1993.

Figure B12-1, Figure B13-1, and Figure B13-2

[Proprietary Information]

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