GE Hitachi Nuclear Energy

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Subject: **Peak Cladding Temperature/10 CFR §50.46 for the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification Renewal Application, Supplemental Information**

During the weeks of August 6 through September 5, 2018, the NRC conducted an audit of the GE Hitachi (GEH) submittals for corrections and errors reported under 10 CFR §50.46 for the Peak Cladding Temperature associated with the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification Renewal Application (Reference 1). This letter submits supplemental information that expands upon the information provided in Reference 2. Specifically, the information in the accompanying Enclosure 1 to this letter re-evaluates the information provided in Reference 3, and describes additional assessments discussed during the audit with the NRC Staff. Enclosure 2 to this letter provides the revised, associated markups for the ABWR Design Control Document (DCD). The proposed changes will be incorporated into Revision 7 of the DCD.

Please contact me if you have questions.

Sincerely,

Michelle P. Catts

Michelle P. Catts
Senior Vice President, Regulatory Affairs

Commitments: None

References:

1. Letter, A. Muñiz to Jennivine Rankin, “Audit Plan for the Advanced Boiling Water Reactor Design Peak Cladding Temperature Increase,” dated July 25, 2018.
2. Letter, J. G. Head (GEH) to Document Control Desk (NRC), M170071, “Peak Cladding Temperature 2016 Annual Reporting Under 10 CFR 50.46 for the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification and the ABWR Design Certification Renewal Application,” dated March 20, 2017.
3. Letter, J. G. Head (GEH) to Document Control Desk (NRC), MFN-16-059, Supplement 1, “Peak Cladding Temperature 2016 Annual Reporting Under 10 CFR 50.46 for the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification and the ABWR Design Certification Renewal Application,” dated October 12, 2016.

Enclosures:

1. GEH Supplemental Information on Peak Cladding Temperature/10 CFR §50.46
2. ABWR DCD Markups, GEH Supplemental Information on Peak Cladding Temperature/ 10 CFR §50.46

cc: A. Muniz, NRC
005N1396

MPC

Enclosure 1

M190008

GEH Supplemental Information

Per 10 CFR § 50.46 Reporting of

Peak Cladding Temperature Changes

Issue:

On February 9, 2017, a public phone call was held in which GEH informed the NRC that GEH would be submitting additional information on the Peak Cladding Temperature (PCT) increase previously described in MFN 16-059 Supplement 1 (Reference 1) and the potential impact on safety analysis. That additional information was provided in a letter, M170071, dated March 20, 2017 (Reference 2). Subsequently, the NRC Staff requested an audit for the purpose of reviewing the changes and errors of the PCT reported by GEH in consideration of the 10 CFR § 50.46 requirements, and to review the additional information reported in that letter (Reference 2). The audit was held via electronic Reading Room and at the GEH offices in Wilmington, North Carolina during the weeks of August 6 through 31, 2018 (Reference 3).

In previous submittals to the NRC, GEH provided PCT adjustments to satisfy the § 50.46 reporting requirement, based upon the changes and errors observed in the operating BWR fleet. No ABWRs have been constructed or operated in the United States since the original DCD certification. Therefore, the original ABWR Emergency Core Cooling System (ECCS) performance method has not been replaced to account for adjustments to the PCT. Because the ABWR is already a certified design, GEH has provided PCT adjustments due to method changes and errors based upon data obtained from operating BWRs with the appropriate corrective adders for the ABWR in accordance with § 50.46 annual reporting requirements. Thus, errors or changes have been conservatively determined and reported.

To test the validity of this method, a comparison was made between the reported changes and errors and the current ABWR design. In a previous GEH submittal (Reference 2), the ABWR standard design was compared with an analysis for a non-domestic ABWR, where the subsequent model incorporated the reported changes and errors. This was done to demonstrate the conservatism in the assessment of the changes and the continued acceptability of the original evaluation model as compared with more modern methods; however, because the ABWR was not originally certified using those methods, the question was presented whether those results would be a valid comparison to the original analysis and methods used.

In that letter, each of the 10 CFR § 50.46 criteria were addressed. This letter will revisit those criteria in light of the original analyses in the DCD and make amendments to forego the use of modern methods in the previous letter. With these amendments, the ABWR standard design remains well within the regulatory criteria. The information, which follows, also addresses the safety analyses that are affected by the PCT adjustments to assure conformity with the regulatory requirements for renewal of a standard design certification, and the DCD is modified for these impacts.

GEH considers the information provided in this Enclosure 1 and, in the DCD, as shown in Enclosure 2, an appropriate update of the ABWR renewal application. GEH believes that the changes discussed in this letter, are in full compliance with the NRC regulations for the renewal of a standard design certification.

GEH Response:

During a public meeting held on the February 9, 2017, GEH proposed to make available to the NRC Staff current ABWR Emergency Core Cooling System (ECCS) analysis results to show that the change in PCT (ΔPCT) provided in MFN 16-059 Supplement 1 (Reference 1), was conservative. This analysis used a modern fuel design and modern analytical methodologies that account for all contributions to the bounding PCT increase described in the letter. In retrospect and after discussions with the NRC Staff during the August 2018 audit, the Reference 2 analysis was problematic in view of the requirement of 10 CFR § 52.63 for finality of the standard design certification for the ABWR. Because of these discussions, GEH determined that a re-evaluation of the ΔPCT information was prudent. The re-evaluation, provided in this Enclosure 1, is based on the methods used in the original, approved DCD submittal.

Background and Summary:

The issues and the comparisons discussed below are associated with 10 CFR § 50.46(a) requirements that were included in the August 28, 2007 amendments for licenses, certifications, and approvals, which focused on alignment with the 10 CFR § 52 process. These amendments added requirements for an applicant for a standard design certification (such as GEH is for the ABWR):

> 10 CFR § 50.46(a)(3)(iii): For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or holder of a standard design approval or the applicant for a standard design certification (including an applicant after the Commission has adopted a final design certification rule) shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission and to any applicant or licensee referencing the design approval or design certification at least annually as specified in § 52.3 of this chapter. If the change or error is significant, the applicant or holder of the design approval or the applicant for the design certification shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with § 50.46 requirements. The affected applicant or holder shall propose immediate steps to demonstrate compliance or bring plant design into compliance with § 50.46 requirements.

GEH has been reporting the § 50.46 information to the NRC at least annually; however, changes were not proposed to be made in the ABWR Design Control Document (DCD) until GEH again reported the information to the NRC in Reference 1. At that time, GEH did not propose a schedule for providing a reanalysis, but as allowed in the regulation, proposed to take "other action as needed to show compliance with § 50.46 requirements." The actions included

providing a note in the ABWR DCD regarding an adjustment to the peak cladding temperature.[1] Also, it was confirmed the DCD requires a plant specific analysis for each licensee citing the DCD to license an ABWR (DCD, Section 6.3.6). This note shows that the calculated maximum fuel element cladding temperature does not exceed 2,200°F, showing compliance to the acceptance criteria. The proposed schedule for providing a re-analysis per § 50.46 requirements when the errors and changes in PCT becomes significant is to use plant specific analyses, as required in Combined License applications. The information below reconfirms compliance with the peak cladding temperature criterion and further demonstrates compliance with the 10 CFR § 50.46 criteria for maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling.

In addition to the requirements for complying with 10 CFR § 50.46, the basis for compliance with the criteria for certification renewal required by 10 CFR § 52.59, is for the design, as originally certified, to comply with the *Atomic Energy Act* and the Commission's regulations applicable and in effect at the time the original certification was issued. Specifically, the changes GEH has proposed to the DCD (1) do not invalidate the original design certification compliance with regulatory requirements, and (2) demonstrate that the changes comply with current 10 CFR § 50.46 requirements. The information presented demonstrates that the ABWR safety analyses are maintained. As discussed, the only safety analyses impacted by the adjustments in the model were the ECCS-Loss-of-Coolant Accident (LOCA) and the Probabilistic Risk Assessment (PRA) ECCS success criteria. The resultant bounding ΔPCT change was 200°F (111°C). This result was revised in the 2018 annual report for the ABWR to be 75°F (42°C) (Reference 4).

The regulatory requirements for features included in an acceptable evaluation model are found in 10 CFR § 50 Appendix K, "ECCS Evaluation Models." Because the ABWR is already a certified design, the evaluation models used at the time of certification continue to be acceptable for this analysis. Further, the previous submittal was confusing as to whether the methods used effected other areas of the plant Safety Analyses. As a result, GEH is submitting this revised discussion of the effect of the PCT changes on the ABWR.

Review of the Additional Safety Analyses:

In the letter, M170071, GEH discussed the cumulative impacts of the model changes reported in the annual reports from 1996 to 2016, which were summarized in Table 5, "Impacts of Model Changes on ABWR Safety Analyses" (Reference 2), modified and shown below as Table 1. In this letter the impacts of the model changes described in M170071 on other ABWR safety analyses are reviewed and re-evaluated using the original methods used in the DCD or engineering analyses, where calculation was not originally used or was unavailable. The results of the re-evaluation are summarized in the following Table 1, which is modified to show

[1] The Reference 1, Enclosure 1 (Note (1) for Table 6.3-4) bounding changes were originally determined in degrees Fahrenheit (*e.g.*, 30°F for the item reported in 1996-01) and each item was converted to degrees Celsius. Because of the rounding of the individual contributions identified, the total sum for the ABWR DCD ΔPCT was reported as 200°F and 111°C. The 200°F and 111°C are not equal to a direct conversion. As a result of the re-evaluation discussed in this letter, those temperatures are determined to be 75°F and 42°C, respectively. (Reference 4)

which items were subsequently determined not to be applicable to the cumulative errors and changes in PCT. Those items marked as “Not Applicable” (NA) are not part of the total cumulative PCT changes and should not be considered in future ABWR licensing bases.

For purposes of clarity, this discussion is divided into two parts: a general discussion of the impacts on the safety analyses due to the adjustments in PCT, and the specific rationale for the elimination of previously reported items from the Table 1 list. The first part includes the general impacts caused by the changes of peak cladding temperature on the overall safety analyses for each of the topics previously identified in M170071. The previous view, and any changes brought about by the re-evaluation identified in the 10 CFR §50.46 annual report for 2018, M180187, (Reference 4) are described below.

Table 1 – Impacts of LOCA Model Changes and Errors on ABWR Safety Analyses

10 CFR §50.46 Reporting Letter	Station Blackout (Appx 1C)	RPV Fluence (5.3)	Decay Heat (5.4)	Containment (6.2)	Combustible Gas (6.2)	ECCS-LOCA (6.3)	Radiological (15)	Transients (15)	ATWS (Appx 15E)	PRA Success Criteria (19)	Applicability
1996-01		P			P	Y		P	P	Y	NA
1992-02						Y				Y	
2001-02						Y				Y	
2001-04						Y				Y	
2002-02						Y				Y	NA
2002-03						Y				Y	
2002-04						Y				Y	NA
2003-01						Y				Y	
2003-03						Y		P	P	Y	
2006-01						Y				Y	NA
2012-01						Y		P	P	Y	NA
2014-01						Y				Y	
2014-02						Y				Y	
2014-03						Y				Y	NA
2014-04						Y				Y	

P - Potential impact with details provided below.
Y - Yes, changes and errors in PCT effect these results the discussion of which follows below.
NA - Further evaluation has determined that this LOCA model change is not applicable to the ABWR DCD.

General Discussion of Impacts

Table 1 identifies the following sections of the DCD that have the potential to have been affected by the changes and errors in PCT.

- Station Blackout – The analysis does not use the same models that are used for the ECCS-LOCA analysis, so there is no impact on the Station Blackout (SBO) analysis due to changes in PCT. Because the SAFER model was not used for the Station Blackout (SBO) analysis in the DCD, any issue associated with the use of SAFER is not relevant for the SBO analysis.

  The original DCD analysis for SBO was based on an engineering evaluation and was not calculated using any code. The ABWR SBO is designed to have an alternate AC power source available to be connected to the equipment providing core inventory and decay heat removal within ten (10) minutes. Therefore, the core is always covered, and the peak cladding temperature is the initial fuel steady state temperature. No coping analysis is required according to 10 CFR § 50.63 rule. Thus, the time-step convergence error noted in 10 CFR § 50.46 report 2001-02 for SAFER does not affect the SBO evaluation.

- Reactor Pressure Vessel (RPV) Fluence – The analysis does not use the same models that are used for the ECCS-LOCA analysis. The incorrect active fuel rod number (LOCA analysis item 1996-01) was the only potential change to the RPV fluence analysis in that annual report. As previously mentioned, no ABWRs have been constructed in the United States. Should an ABWR be constructed, a plant specific RPV fluence analysis would be performed in accordance with the guidance of U.S. NRC Regulatory Guide 1.190 (Reference 5). That plant specific analysis would reflect the best estimate of RPV fluence accrual and account for the intended core loading and operating strategy through the end of plant life. Thus, any variations in the fuel lattice or bulk cladding temperature from the observations noted above would be accounted for at that time.

  RPV fluence analyses are performed at standard operating conditions for nuclear reactors and are intended to be a best estimate for a given plant. The RPV fluence analysis does not rely on the SAFER models that are used for the ECCS-LOCA analysis. Thus, an increase in PCT due to the accident condition analyzed by the ECCS-LOCA analysis does not impact the RPV fluence assessment. Therefore, the change reported in the 1996-01 annual report of 10 CFR §50.46 changes, is not relevant to the ABWR and was eliminated from this discussion.

- Decay Heat – The analysis does not use the same models that are used for LOCA analysis. The decay heat evaluations are inputs into the ECCS and other safety analyses. The MFN 16-059 Supplement 1 listing of required changes to ECCS analysis does not include any items that would impact the decay heat results. The analysis that supports the ABWR DCD was based on the approved licensing basis at the time of certification.

- Containment – The analysis was updated in ABWR DCD Rev 5, which was the basis for the ABWR design certification renewal application. The analysis does not use the same models that are used for the ECCS-LOCA analysis.

- Combustible Gas – The analysis is based on the requirements of Regulatory Guide 1.7(Reference 6). The analysis does not use the same models that are used for the ECCS-LOCA analysis. The fuel bundle design is a primary input to the evaluation. The ECCS-LOCA analysis item 1996-01 is the only potential change to the evaluation and further review determined that the incorrect fuel rod number was in the SAFER analysis only.
- ECCS-LOCA – See discussion, which follows below.
- Radiological - The analysis does not use the same models that are used for ECCS-LOCA analysis except for the break flows and the associated mass releases. There were no required model changes associated with the break flow calculations. The analysis is not dependent upon fuel type but rather mass of core and exposures.
- Transients and ATWS –Table 1 shows that only the separator pressure drop item (LOCA model error item 2003-03) potentially effects transients and ATWS and is applicable to the ABWR DCD. The ABWR DCD Transient and ATWS analyses base decks are developed separately from the LOCA analysis. The ABWR DCD Transient and ATWS analyses base decks used the correct separator pressure drop.

  In M170071 there were three LOCA model changes and errors that had the potential to affect ABWR DCD transient and ATWS. Two of those items (1996-01 and 2012-01) were determined not to be applicable to the ABWR DCD.

  - The incorrect active fuel rod number (LOCA model error item 1996-01) was determined to not be applicable to the ABWR DCD. There is no impact on ABWR DCD transient or ATWS analysis.
  - The effects of fuel Thermal Conductivity Degradation (TCD) (LOCA model change 2012-01) was determined to not be applicable to the ABWR DCD; however, the effects of fuel TCD are not included in the ABWR DCD transient and ATWS analysis. A discussion of the effect of fuel TCD on ABWR DCD transient and ATWS analysis is provided below.

    The ABWR DCD Transient and ATWS analyses are not significantly affected by fuel TCD. This is based upon GEH experience and was previously evaluated by the U.S. NRC in Reference 7. Reference 7 provides an in-depth assessment of GEH and GNF codes and methods with regard to fuel TCD. The codes and methods evaluated in Reference 7 are the same codes and methods used in the ABWR DCD transient and ATWS analyses. The conclusions of the U.S. NRC evaluation were largely based on sensitivities provided in Reference 8 (RAI 39) and Reference 9.

    The conclusions for transients were based on sensitivities for a BWR/4. The BWR/4 transient is very similar to the ABWR transient in both sequence of the event and importance of phenomena; therefore, the conclusions of the sensitivities are applicable to the ABWR DCD analysis. Additionally, the ABWR DCD core is an initial core with relatively low exposure (only fresh bundles);

therefore, the ABWR DCD analysis would be expected to have lower sensitivity to fuel TCD than the BWR/4 sensitivities. The conclusions for transients are also applicable to the trip of all reactor internal pumps (which is transient event with special acceptance criteria in the ABWR DCD). Additionally, for the trip of all reactor internal pumps, the event is limiting at beginning of cycle of the initial core, due to the low void reactivity feedback. This is a condition with all fresh fuel; therefore, there is no effect of fuel TCD.

The conclusions for ATWS were based largely on sensitivities for the ESBWR. Although the analysis used is based on the ESBWR, the ATWS scenario and phases of the event and importance of phenomena are very similar between the ESBWR and the ABWR ATWS analysis; therefore, the conclusion is applicable to the ABWR DCD ATWS analysis.

- PRA - The PRA calculations for LOCA events use ECCS success criteria that are developed using the ECCS-LOCA models to determine the minimum amount of ECCS systems/capacity that is needed for successful event mitigation. To be conservative, the 75°F (42°C) adder has been introduced to the reported values in the DCD (See Enclosure 2). There is no impact on the conclusion of the evaluation. The remaining PRA evaluations were performed using the MAAP code, which it is independent of the ECCS-LOCA models.

Specific Rationale for Elimination of Entries from Table 1

ECCS-LOCA – The letter, previously submitted, M170071 (Reference 2), discussed the impacts on the ECCS-LOCA analyses. These impacts have been modified in the section below that discusses the specific rationale for the elimination of certain adjustments.

As a second part of the discussion, some of the entries in Table 1 were determined to be Not Applicable (NA) to the cumulative PCT changes and errors. Each item will be referred by its respective letter number identified in Table 1.

1. Letter 1996-01: This item was reported for the ABWR ECCS-LOCA analysis as a precautionary finding, not confident of the timing when this erratum was discovered, how long it would have been extant in the input generation tool, and whether such would have been impactful at the time of the ABWR DCD analysis. Subsequent study into archived records has confirmed the fuel bundle analyzed for ABWR would not be affected. The item is reported to the NRC (MFN-088-96) as applicable for bundles with large water holes. The fuel bundle analyzed for ABWR is confirmed not to be characterized as such, designed with two, typically-sized water holes (ABWR Systems Safety Analysis Report Table 1.3-1). As such this item was removed from the summary for the ABWR DCD.
2. Letter 2002-02: This letter was related to steam dryer pressure drop. In reviewing this issue, it was determined that the historic design file contains a preliminary calculation to discern the pressure drop across the steam dryer for ABWR. This defined input which was then introduced into the ABWR design calculations to account for the difference in pressure across the dryer. So, the analysis basis was performed correctly on this point.

The reporting for 2002-02 was applied to analyses across the fleet where this determination was not explicit. Since the ABWR was not included in the survey of operating plants for this notification, it was not checked at the time of issuance. Upon this further review - concluding it was modelled accurately - it would be screened out from the list of plants for which the item would apply. Because this evaluation did use the system model to identify a plant-specific value for this input, in contrast to many of the rest of the BWR fleet, the Notification Letter 2002-02 would not be applicable to the ABWR analysis. As a result, it may be removed from the total cumulative PCT changes and errors.

3. Letter 2002-04: This item was a pro forma reporting to acknowledge porting of the evaluation model software to an alternate computational platform, demonstrated to have no impact on results. As with other, comparable homes for the software, the direction of the certification renewal is to support the analysis on the basis of the evaluation model used at the time of the analysis. As such this and reporting of similar residence action is not required, would not be impactful on the analysis as reported, and are removed.

4. Letter 2006-01: The original assessment applied a course sensitivity based on BWR sensitivities, which, upon discussion and review, was found to be inappropriate. The Steam Line Outside Containment is the limiting ABWR break location. The combination of the rapid depressurization and the trip of all reactor internal pumps with subsequent fast reduction in core flow results in a short duration departure from nucleate boiling (DNB) at the initiation of the event. This is not a core uncovery, as was previously projected, and upon which this item was based when reported. The heat up from the DNB is of very short duration with a relatively low temperature excursion to a peak of 1,149ºF within 10 seconds into the event.

   Of greater significance to the ABWR, the depressurization results in liquid entrainment that rapidly cools the fuel back to saturation conditions. This same phenomenon of DNB and immediate cooling would be expected as response for this analysis irrespective of the initial power shape. This is much different for ABWR than the phenomena being addressed by 2006-01 where there is a long-term core uncovery for a small break LOCA scenario; a scenario where a more top peaked shape could have greater impact. Therefore, the effect of top-peaked power shape is concluded to show no impact on the ECCS-LOCA analysis for ABWR; Notification Letter 2006-01 would not be applicable to the ABWR analysis.

5. Letter 2012-01: This report was issued with NRC concurrence (IMLTR, NEDC-33173P, Supplement 4), to implement a resolution to considerations related to the Fuel Thermal Conductivity Degradation (TCD) issue. Notwithstanding this action, it was seen that no added method or model would be required to disposition TCD for ABWR, so the letter was removed from the list as not applicable for the DCD.

   With respect to TCD, itself, GEH has previously presented the finding that TCD becomes relevant at later exposure times of fuel residence in the core. The ECCS-LOCA analysis is bounding at early exposure times (as a function of gap conductance, stored energy). The ABWR ECCS-LOCA analysis, performed at a bounding early time

in life, would not be appreciably affected by TCD. This conclusion is consistent with the assessment that was documented by the NRC in Reference 7, based on the GESTRM evaluation model, the same model as was used for the analysis reported in the DCD. As a result, the original evaluation model, which was contemporary with the certification, can be relied upon for PCT analysis. All reference to PRIME has been removed from the next DCD revision, as shown in Enclosure 2. The ECCS-LOCA analysis for ABWR would remain valid based on methodology bounding at early time of fuel exposure, and TCD would not challenge that result for the analysis of the DCD.

6. Letter 2014-03: The issues presented in this report dealt with modeling during low flow conditions. In that situation, liquid droplets predicted above the two-phase level in the core would fall and potentially lead to overpredicting void volume and a lower core pressure drop than would be accurate. Consequence of this pressure condition would be increased leakage backflow from bypass leading to non-conservative cooling effects. The solution was to impose a minimum core pressure difference in the calculation. The model change reported by 2014-03 was removal of this minimum core pressure difference and correction of the pressure calculation scheme for low flow conditions.

   This item was originally reported for the ABWR summary, expecting this code change to be in the model at the time of performance of the ABWR ECCS-LOCA analysis. However, as the time line was laid out for this model change, it was seen the ABWR analysis preceded imposition of the minimum pressure difference, via the updated model version, by some number of months. The analysis would not be affected.

   The original analysis was performed with the prior SAFER03 version. Because SAFER04E4 was not used, the change reported in the 2014-03 letter would not be applicable to the ABWR analysis. As a result, the changes identified in PCT Notification Letter 2014-03, can be removed from the ABWR cumulative listing.

For clarity, the remaining reports are identified in Table 2, “Impacts of Model Changes on ABWR Safety Analyses based on Re-evaluation of Peak Cladding Temperature,” which follows. These are the factors for which the new evaluation, based upon the existing methods used in the approved DCD and engineering analyses were used. These form the basis for the most recent annual reporting of Reference 4. The new total cumulative PCT value of 1,224°F will form the licensing basis for PCT in future licensing applications.

Table 2 – Impacts of Model Changes on ABWR Safety Analyses based on Re-evaluation of Peak Cladding Temperature

Annual Report Description	LOCA (6.3)	Transients (15)	ATWS (Appx 15E)	PRA Success Criteria (19)	Calculated PCT value
Original Report	Y				1,149 °F
1992-02 CCFL in upper tie plate, pre-GE-11	Y			Y	+25 °F
2001-02 Time step change for convergence	Y			Y	+ 25 °F
2001-04 Steam condensed by ECCS injection	Y			Y	+ 10 °F
2002-03 GESTR input file interpolation	Y			Y	0 °F
2003-01 SAFER Level Volume Table	Y			Y	+ 10 °F
2003-03 Steam Separator Pressure Drop	Y			Y	+ 5 °F
2014-01 Code Changes of Neutral Effect	Y			Y	0 °F
2014-02 Mass Non-Conservatism	Y			Y	0 °F
2014-04 Lower Plenum CCFL	Y			Y	0 °F
Cumulative and Absolute Sum of 10 CFR § 50.46 Changes					75 °F
Projected Licensing Basis PCT Values					1,224 °F

Y - Yes, changes and errors in PCT effect these results, the discussion of which is above.

LOCA Maximum Oxidation and Hydrogen Generation Analyses

Although the cumulative changes and errors reported to the evaluation model have resulted in changes to peak cladding temperature, they have not significantly affected the other 10 CFR § 50.46 criteria. The maximum cladding oxidation reported for the ABWR remains appreciably unchanged, from that shown in the DCD due to these changes and errors. Differences in these values, if any, are in the thousandths of percent and are effectively eliminated when rounding these numbers to the significant figures shown. The ABWR maintains significant margin below the allowable limit of 17%. The calculated total hydrogen generated from the chemical reaction of the cladding with the water or steam remains unchanged from that shown in the original DCD. Again, a substantial margin is maintained to the 1% core wide limit.

For ABWR, there is no core uncovery. Hence, a coolable core geometry is maintained. Long-term core cooling is ensured by using redundant reactor heat removal systems with adequate water sources to keep the core covered and the transfer of decay heat in the core to the ultimate heat sink. Therefore, no reportable effect has been noted in either the LOCA maximum oxidation or the hydrogen generation reported in the DCD.

Conclusion of the Impact of 10 CFR § 50.46 Items on DCD Safety Analyses:

In conclusion, the only affected safety analyses in the ABWR DCD were the ECCS-LOCA and the PRA ECCS success criteria and the resultant bounding change on PCT was 75°F (42°C). The total of the cumulative errors and changes in peak cladding temperature reported in the revised 10 CFR § 50.46 annual report (Reference 4) are seen to potentially affect the calculated results based on the evaluation model as applied in the analysis of the DCD. It is concluded that the existing evaluation model for the ABWR DCD fuel type remains valid. These cumulative errors and changes in peak cladding temperature, provided in Enclosure 2, will be incorporated in the next revision of the DCD.

With the next DCD revision, the ABWR DCD will be modified to account for the total cumulative impact identified for peak cladding temperature. No additional DCD changes are necessary to address adjustments to the ECCS model which were identified since the original design certification.

References:

1. Letter, J. Head (GEH) to F. Akstulewicz (NRC), "Peak Cladding Temperature 2016 Annual Reporting Under 10 CFR 50.46 for the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification and the ABWR Design Certification Renewal Application," dated October 12, 2016 (MFN-16-059, Supplement 1).
2. Letter, J. Head (GEH) to NRC, "Peak Cladding Temperature/ 10 CFR 50.46 2016 for the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification Renewal Application, Supplemental Information," dated March 20, 2017 (M170071).
3. Letter, A. Muñiz to Jennivine Rankin, "Audit Plan for the Advanced Boiling Water Reactor Design Peak Cladding Temperature Increase," dated July 25, 2018.
4. Letter, M. Catts (GEH) to Document Control Desk (NRC), "Summary of Changes and Errors in ECCS Evaluation Models," dated October 29, 2018. (M180187).
5. Reg Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001.
6. Reg Guide 1.7, "Control of Combustible Gas Concentrations in Containment," dated March 2007.
7. Adams Accession Number: ML120750001, NRC Staff Assessment of GE-Hitachi Nuclear Energy / Global Nuclear Fuel - Americas Codes and Methods with Regard to Thermal Conductivity Degradation, dated March 23, 2012 [https://www.nrc.gov/docs/ML1207/ML120750001.pdf].
8. Global Nuclear Fuel, NEDC-33256P, Revision 1, The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 1—Technical Bases, dated September 2010.
9. Enclosure 1 of MFN 08-713, Letter, Richard E. Kingston (GEH) to U.S. Nuclear Regulatory Commission Document Control Desk, Response to Portion of NRC Request for Additional Information Letter No. 156 Related to ESBWR Design Certification Application - Emergency Core Cooling Systems - RAI Number 6.3-54 S01, dated September 22, 2008.

Enclosure 2

M190008

ABWR DCD Redline

Supported by

GEH Supplemental Information

on

Peak Cladding Temperature/ 10 CFR § 50.46

Table 1.6-1 Referenced Reports (Continued

References shown as being added in M170071 have been removed from this DCD markup. This note will not appear in the revised DCD.

Report No.	Title	
NEDC-30851P-A	W. P. Sullivan, "Technical Specification Improvement Analyses BWR Reactor Protection System", March 1988.	
NEDE-31096-A	"GE Licensing Topical Report ATWS Response to NRC ATWS Rule 10 CFR 50.62", February 1987.	19B.2
NEDE-31152-P	"GE Bundle Designs", December 1988.	4.2
NEDO-31331	Gerry Burnette, "BWR Owner's Group Emergency Procedure Guidelines", March 1987	18A
NEDC-31336	Julie Leong, "General Electric Instrument Setpoint Methodology", October 1986.	7.3
NEDC-31393	"ABWR Containment Horizontal Vent Confirmatory Test, Part I", March 1987.	3B
NEDO-31439	C. VonDamm, "The Nuclear Measurement Analysis & Control Wide Range Neutron Monitoring system (NUMAC-WRNMS)", May 1987	20.3
NEDC-31858P	Louis Lee, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control System", 1991	15.6
NEDE-31906-P	A. Chung, "Laguna Verde Unit I Reactor Internals Vibration Measurement", January 1991.	7.4
NEDO-31960	Glen Watford, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology", June 1991.	4.4
NEDC-32267P	"ABWR Project Application Engineering Organization and Procedures Manual", December 1993.	17.1
NEDO-32686-A	"Utility Resolution Guide for ECCS Suction Strainer Blockage", October 1998.	6C

Because either the ADS initiating signal, or the overpressure signal opens the safety-relief valve, no conflict exists.

The LPFL Subsystem is configured from the RHR pumps and some of the RHR valves and piping. When the reactor water level is low, the LPFL Subsystem (line up) has priority through the valve control logic over the other RHR Subsystems for containment cooling. Immediately following a LOCA, the RHR System is directed to the LPFL mode. When the RHR shutdown cooling mode is utilized, the transfer to the LPFL mode must be remote manually initiated.

References shown as being added in M170071 have been removed from this DCD markup. This note will not appear in the revised DCD.

6.3.3.6 Limits on ECCS Parameters

Limits on ECCS parameters are given in the sections and tables ref Subsections 6.3.3.1 and 6.3.3.7.1. Any number of components in a of service, up to the entire system. The maximum allowable out-of of the level of redundancy and the specified test intervals.

6.3.3.7 ECCS Analyses for LOCA

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

The methods used in the analysis have been approved by the NRC or meet the change criterion in 10CFR50.46. For the system response analysis, the LAMB/SCAT and SAFER/GESTR models approved by the NRC were used. The significant input variables used for the response analysis are listed in Table 6.3-1 and Figure 6.3-44.

6.3.3.7.2 Accident Description

The operation sequence of events for the limiting case is shown in Table 6.3-2.

6.3.3.7.3 Break Spectrum Calculations

A complete spectrum of postulated break sizes and locations were evaluated to demonstrate ECCS performance. For ease of reference, a summary of figures presented in Subsection 6.3.3.7 is shown in Table 6.3-5.

A summary of results of the break spectrum calculations is shown in tabular form in Table 6.3-4 and graphically in Figure 6.3-10. Conformance to the acceptance criteria (PCT=1204°C, Local oxidation = 17% and core-wide metal-water reaction = 1%) is demonstrated for the core loading in Figure 4.3-1. Results for the limiting break for each bundle design in a plant will be given for information to the USNRC by the COL applicant. See Subsection 6.3.6.3 for COL license information. Details of calculations for specific breaks are included in subsequent paragraphs.

6.3.3.7.4 Large Line Breaks Inside Containment

Since the ABWR design has no recirculation lines, the maximum steamline break (985 cm^2), Maximum feedwater lines break (839 cm^2), and the maximum RHR shutdown suction line

All instrumentation required for automatic and manual initiation of the HPCF, RCIC, RHR and ADS Systems is discussed in Subsection 7.3.1, and is designed to meet the requirements of IEEE-279 and other applicable regulatory requirements. The HPCF, RCIC, RHR and ADS Systems can be manually initiated from the control room.

The RCIC, HPCF, and RHR Systems are automatically initiated on low reactor water level or high drywell pressure. The ADS is automatically actuated by sensed variables for reactor vessel low water level and drywell high pressure plus indication that at least one RHR or HPCF pump is operating. The HPCF, RCIC, and RHR Systems automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. The RHR LPFL mode injection into the RPV begins when reactor pressure decreases to the RHR's pump discharge shutoff pressure.

HPCF injection begins as soon as the HPCF pump is up to speed and the injection valve is open, since the HPCF System is capable of injection water into the RPV over a pressure range from 8.12 to 0.69 MPaD or pressure difference between the vessel and drywell.

References to Prime shown as being added in M170071 have been removed from this DCD markup. This note will not appear in the revised DCD.

6.3.6 COL License Information

6.3.6.1 ECCS Performance Results

The exposure-dependent MAPLHGR, peak cladding temperature, and oxi each fuel bundle design based on the limiting break size will be provided to the USNRC for information (Subsection 6.3.3).

6.3.6.2 ECCS Testing Requirements

In accordance with the Technical Specifications, the COL applicant will perform a test every refueling in which each ECCS subsystem is actuated through the emergency operating sequence (Subsection 6.3.4.1).

6.3.6.3 Limiting Break Results

Results for the limiting break for each bundle design will be provided to the USNRC by the COL applicant (Subsection 6.3.3.7.3).

6.3.7 Reference

6.3-1 "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K", (NEDE-20566-P-1), September 1986.

Add: 6.3-2 M180187, "Peak Cladding Temperature 2018 Annual Reporting Under 10 CFR §50.46 for the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification and the ABWR Design Certification Renewal Application" (October 29, 2018), Enclosure 1, "Advanced Boiling Water Reactor 2018 Annual Report Under 10 CFR § 50.46(a)(3)(iii)."

Table 6.3-4 Summary of Results of LOCA Analysis including Renewal PCT with 10 CFR § 50.46 Adjustments

Break Location	Break Size[1] (cm^2)	Systems Available	PCT (°C)	Renewal PCT/ w ΔPCT adjustment (°C)[2]	Maximum Local Oxidation[3]
Based on Appendix K evaluation models:					
Steamline Inside Containment	985	1HPCF + RCIC +2 RHR/LPFL + 8 ADS	552	594	0.03%
Feedwater Line	839	1 HPCF + 2 RHR/LPFL + 8 ADS	542	584	0.03%
RHR Shutdown Cooling Suction Line	792	1 HPCF + RCIC + 2 RHR/LPFL + 8 ADS	542	584	0.03%
RHR/LPFL Injection Line	205	1 HPCF + RCIC + 1 RHR/LPFL + 8 ADS	542	584	0.03%
High Pressure Core Flooder	92	RCIC+2 RHR/ LPFL + 8 ADS	542	584	0.03%
Bottom Head Drain Line	20.3	1HPCF + RCIC + 2 RHR/LPFL + 8 ADS	542	584	0.03%
Steamline Outside Containment	3939	1 HPCF + RCIC + 2 RHR/LPFL + 8 ADS	621	663	0.03%
Based on bounding values:					
Steamline Outside Containment	3939	1 HPCF + RCIC + 2 RHR/LPFL + 8 ADS	619	661	0.03%

[1] The most severe ABWR design basis LOCA calculations (Subsection 6.3.3.7.8) involve use of bounding worst-case values for key plant parameters - including an arbitrary 20% increase in the break flow rate. Even with these bounding assumptions, the LOCA analyses demonstrate that the ABWR design still retains large margins between predicted peak fuel clad temperatures and the criteria of 10 CFR 50, Appendix K. Tolerances associated with fabrication and installation may result as-built break areas that could be 5% greater than these values. Based on the above conservatisms in the LOCA analyses, these as-built variations would not invalidate the plant safety analyses presented in Chapter 6 and in Chapter 15.

[2] The cumulative ΔPCT result of 42°C was added to each of the calculated PCT values to account for errors and changes to the evaluation model that have been reported since the DCD was originally approved. These new values form the renewal licensing basis for future licensing applications. The estimated adjustment of the PCT is based on reporting under 10 CFR § 50.46 and is a potential increase of 42°C, when added to the calculated values shown, this result demonstrates continued compliance to the 1,204 °C (2,200 °F) regulatory acceptance criteria limit with ample margin. See Reference 6.3-2. [This is Former Note 2, shown in M170071, the comment in these square parentheses will be removed in the DCD].

[3] The core-wide metal-water reaction for this analysis has been calculated using method 1 described in Reference 6.3-1. This results in a core-wide metal-water reaction of 0.03%. [This is Former Note 1, shown in M170071, the comment in these square parentheses will be removed in the DCD].

Section 19.3.1.3.1

condensate storage tank. Sufficient makeup water is available to enable these pumps to maintain adequate core cooling for all events except large or medium liquid LOCAs.

A motor driven feedwater pump is combined in series with a condensate pump in order to provide a higher pressure system. Therefore, this option also depends on the availability of makeup water and electrical power. Sufficient makeup water is available to enable this series of pumps to maintain adequate core cooling for the small steam LOCA and transient events.

The fire protection system has two pumps which take suction from the firewater tanks and inject into the RPV through an RHR line. One pump is driven by an electric motor which requires AC power. The other is driven directly by a diesel engine. Once the reactor system has been depressurized, either pump can provide enough makeup water to restore and maintain the RPV water level following any transient (including IORV) event. The analysis to support this conclusion assumes a full ADS blowdown begins within 15 minutes after the vessel water level has reached the level 1 setpoint. The subsequent reactor system depressurization allows injection from the fire protection system about 7 minutes after the start of the blowdown. The ability of the fire protection system to mitigate the consequences of LOCA events is conservatively ignored. For more information about the fire protection system refer to Subsection 5.4.7.

All calculated PCTs in 19.3 were conservatively updated to account for the estimated adjustments based on 10 CFR §50.46. See Reference 19.3-8.

It is conservative to use the 1204°C (2200 °F) PCT licensing limit as an acceptance criterion for the success criteria since tests have been performed which show that the core will remain in a coolable geometry with temperatures as high as 1482°C (2700°F).

A review of Table 19.3-2 shows that, for success, the inventory threatening events require the flow equivalent of only 1 RHR/LPFL or 1 HPCF pump available for large break cases and only 1 HPCF or 1 RHR/LPFL + 3 ADS available for small break cases. The resulting PCTs for the large break cases and transients were between ~~482°C (900°F) and 593°C (1100°F).~~ For the small break cases with the flow equivalent of only 1 HPCF available the resulting PCTs were less than ~~538°C~~ and with 1 RHR/LPFL + 3 ADS available the maximum PCT was ~~982°C (1800°F).~~

524°C (975°F) and 635°C(1175°F)

580°C

1024°C (1875°F)

1024°C

Subsection 6.3.3.7.8 identifies the input parameters that significantly impact the LOCA results. If the above analyses were reanalyzed with these conservative input parameters, it is estimated that only the resulting PCTs for the small break cases with 1 RHR/LPFL + 3 ADS available are above ~~982°C (1800°F)~~. For these cases the PCT is estimated to be about ~~1260°C (2300°F).~~ However, even for these conservative LOCA calculations all the PCTs are less

(1875°F)

1302°C (2375°F)

19.3.5 References

19.3-1 "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessment," NUREG/CR-3862, Idaho National Engineering Laboratory, May 1985.

19.3-2 "Advanced Light Water Reactor Utility Requirements Document, Volume II, Chapter 1; Appendix A: PRA Key Assumptions and Groundrules," Draft, Electric Power Research Institute, August 1988, p. D4.

19.3-3 "GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment," 22A7007, General Electric Company, March 1982.

19.3-4 "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, NUREG-75/014, United States Atomic Energy Commission, October 1975.

19.3-5 "Failure Rate Data Manual for GE BWR Components", NEDE-22056, Rev. 2, Class III, General Electric Company, January 17, 1986.

19.3-6 A.D. Swain and H.E. Guttmann, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications", NUREG/CR-1278, August 1983.

19.3-7 "Analysis of a High Pressure ATWS with Very Low Makeup Flow", DOE/ID-10211, Idaho National Engineering Laboratory, October 1988.

ADD: 19.3-8 M180187, "Peak Cladding Temperature 2018 Annual Reporting Under 10 CFR 50.46 for the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification and the ABWR Design Certification Renewal Application" (October 29, 2018), Enclosure 1, "Advanced Boiling Water Reactor 2018 Annual Report Under 10 CFR 50.46(a)(3)(iii)."