Issue:

In the February 9, 2017 public phone call between the NRC and GEH, 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 additional safety analysis. The response below provides that information.

In previous submittals to the NRC, GEH has described PCT adjustments due to the 50.46 method changes and errors based on operating BWRs. No domestic ABWRs have been constructed and operated, so the original ABWR ECCS performance method has not been replaced to account for these adjustments. To demonstrate that the ABWR standard design ECCS performance remains within the regulatory limits, GEH has compared the ABWR standard design to analyses that have been updated for a non-domestic ABWR. This is in lieu of adjusting the earlier codes and methods, which, as discussed in public teleconferences, is not practical, considering the evolution of computer capabilities and methods that exist today. As described below, the analyses represent the ABWR standard design with minor differences (which are explained) and predict the results that would be achieved if the ABWR standard design were to be reanalyzed, and address each of the five criteria in 10 CFR 50.46. In addition, the updated codes are being added to the ABWR DCD to reflect that they are now part of the design basis information for the renewed ABWR.

As described below, GEH has used an analysis for a non-domestic ABWR that addresses the methodology adjustments that have been reported for the ABWR. The adjustments in the peak cladding temperature that are made in the DCD markups shows that the regulatory limits are met. As stated above, the 10 CFR 50.46 criteria are addressed below as well and it is concluded that the ABWR standard design remains well within the regulatory requirements. The information below also addresses the safety analyses that are impacted by the PCT adjustments to assure conformation with the regulatory requirements for renewal of a standard design certification and the DCD is modified for these impacts.

GEH considers that, with the information provided below and in the DCD, the ABWR renewal application is updated appropriately and is in compliance with NRC regulations for the renewal of a standard 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, October 12, 2016 (MFN-16-059, Supplement 1).
2. Letter, J. Head to USNRC, 10 CFR 50.46 Annual Report for the GE ABWR Standard Plant Design -2013, Dated December 13, 2013 (MFN 13-095)

GEH Response:

During the February 9, 2017, public meeting, GEH proposed to make available to the 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) is conservative. This analysis uses a modern fuel design and modern methodologies that account for all contributions to the bounding PCT increase described in the letter.

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 Part 52 process. These amendments added requirements for an applicant for a standard design certification (such as GEH[1] 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 first reported the information to the NRC in Reference 2 in 2013. At that time an evaluation was completed but no changes were made in the ABWR Design Control Document (DCD). This information continued to be reported in subsequent reports in 2014, 2015 and 2016. When GEH again reported the information to the NRC in Reference 1, it did not propose a schedule for providing a reanalysis, but as allowed in the regulation, took other action as needed to show compliance with § 50.46 requirements. The actions included providing a note in the the ABWR DCD regarding an adjustment to the peak cladding temperature (Note (1) for Table 6.3-4). This note shows that the calculated maximum fuel element cladding temperature, with the conservative adjustments in the report included in Reference 1, does not exceed 2200°F. The information

[1] GE Hitachi Nuclear Energy is the successor for GE Nuclear Energy, which was the entity for the original ABWR design certification application. GE Hitachi Nuclear Energy is currently the applicant for a renewal application for the ABWR design certification.

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 information below provides the basis for compliance with the criteria for renewal in 10 CFR 52.59 in that the design, as originally certified or as modified, complies with the Atomic Energy Act and the Commission's regulations applicable and in effect at the time the 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 below demonstrates that the ABWR safety analyses are maintained. As discussed below, the only safety analyses impacted by the adjustments in the model were the ECCS-LOCA and the PRA ECCS success criteria and the resultant bounding ΔPCT change was 200°F (114°C[2]).

Modern Methodology:

The discussion below demonstrates that the ABWR renewal application contains, with the proposed DCD changes, the information necessary to bring up to date the information and data contained in the previous application that has been identified (1) regarding the reported errors or changes in the acceptable evaluation model or in the application of such a model that affects the temperature calculation, as required by 10 CFR 50.46, and (2) as impacts on other sections of the DCD due to the increase in the peak cladding temperature.

As described in the ABWR DCD the Loss of Coolant Accident (LOCA) analysis was performed with the NRC-approved LAMB/SCAT and SAFER/GESTR models. The current ABWR ECCS analysis with modern fuel and methodologies was performed with the NRC-approved LAMB/TASC and SAFER/PRIME models. Figure 1 shows the relationship between the codes. Note CORCL is only used in the United States in BWR/2 plants.

[2] Note- The Reference 1, Enclosure 1 bounding changes were originally determined in degrees Fahrenheit (e.g. the 30°F for item 1996-01) and each item was then converted to degrees Celsius. Because of the rounding of the individual contributors, the total sum for the ABWR DCD ΔPCT is 200°F and 114°. The 200°F and 114°C are not equal to a direct conversion.

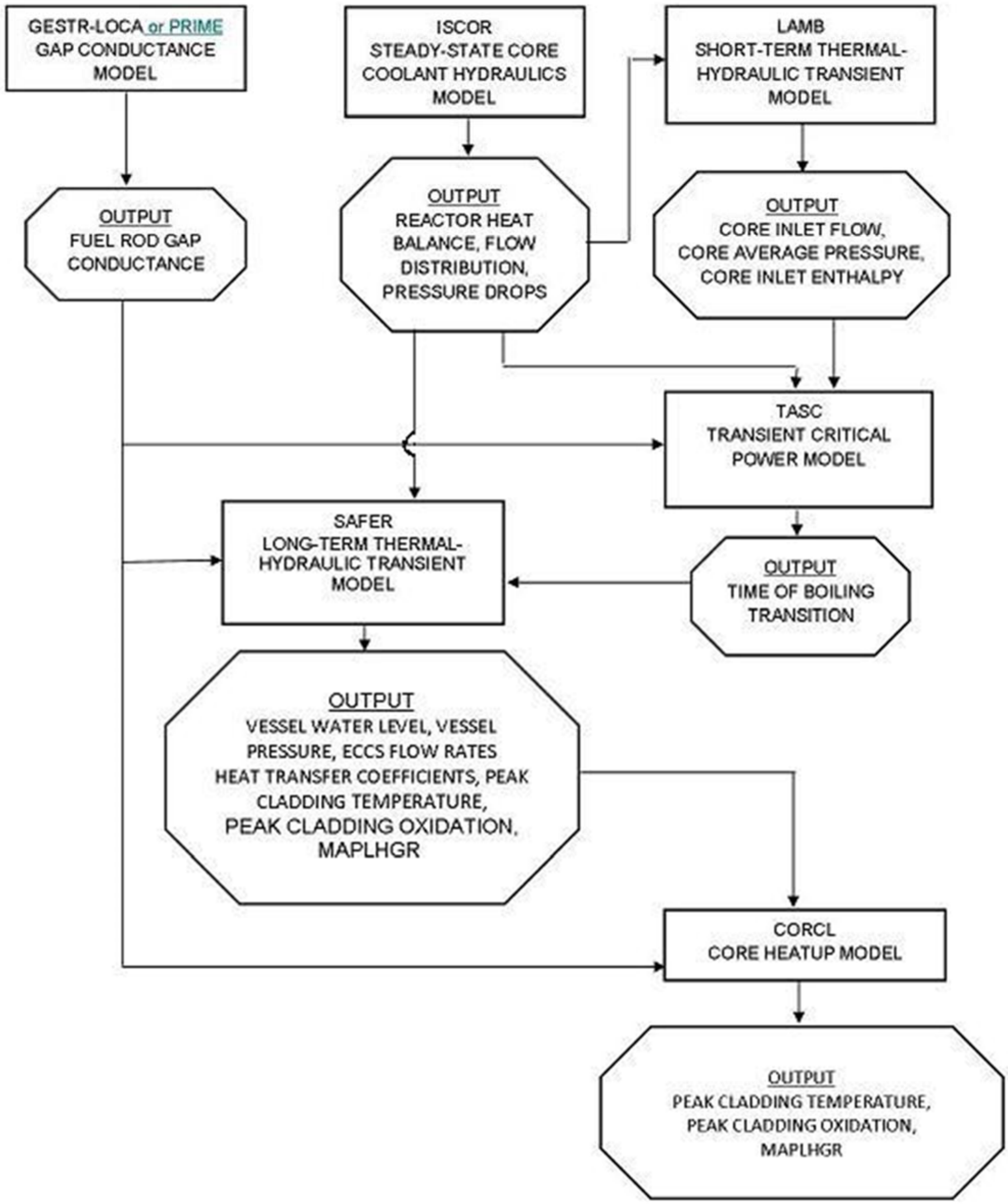

Figure 1 – Relationship Between Codes

The impacts of using different models on the analyses results are described below. The different models are as follows (original model is listed first):

- SCAT and TASC
- GESTR and PRIME

The following is an extract from the USNRC Safety Evaluation for approval of the use of TASC. It is contained in *TASC-03A, A Computer Program for Transient Analysis of a Single Channel,* NEDC-32084P-A Revision 2:

> The staff does agree that TASC is a functional replacement of SCAT. Furthermore, TASC is applied to small parts of a lengthy, complicated analyses. Given this fact and the fact that SCAT and TASC are similar, the staff review of TASC focused on its peformance relative to applicable test data.
>
> TASC is used to predict early boiling transition in LOCA analysis. This typically occurs with the first several seconds of the blow-down for a large break LOCA. This value is used in subsequent codes to turn off nucleate boiling heat transfer models and turn on transition boiling models. The equations and constitutive relationships in the code model the relevant physics necessary to predict this phenomena. The assessment against both TLTA and Atlas data presented in Chapter 4 of the topical report demonstrates that the equations in TASC are solved with sufficient accuracy to be used in licensing applications. The results of the assessment demonstrate that on average TASC conservatively predicts early boiling transition within one second. This level of accuracy is acceptable because the conditions at dryout are not changing rapidly enough within one second to significantly alter the post dryout behaviour. This provides reasonable assurance that TASC can be used to predict early boiling transition for LOCA analysis.

As described in the USNRC staff's Final Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report NEDC-33256P, NEDC-33257P, and NEDC-33258P "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance" (ML100150653), the use of PRIME is not expected to have a significant impact on PCT.

A process was proposed and supported by the NRC for implementation of the PRIME model to the ECCS-LOCA analysis (Supplement 4, NEDO-33173P[3]) which directed completion of the 10 CFR 50.46 reporting noted in the foregoing by assessment of a potential (conservative) estimate of PCT effect based on stored energy and reporting this as a change to the evaluation model, per the regulation.

The NRC also audited the implementation of the PRIME models and data in the downstream methods and concluded that:

> The NRC staff's audit of GEH's PRIME implementation into downstream safety analysis analytical methods found that the NEDO-33173, Supplement 4 plan was correctly

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[3] Implementation of PRIME Models and Data in Downstream Methods, NEDO-33173 Supplement 4-A, Revision 1, dated November 2012.

executed. The PRIME conductivity models were correctly encoded into downstream applications and test cases demonstrated that the impact of switching from GESTRM to PRIME models was as expected.[4]

In summary, the LOCA model changes between the ABWR DCD and the models used for this current ABWR ECCS analysis have been approved by the NRC and the impacts on PCT are bounded by the estimates provided in the 10 CFR 50.46 report provided in the MFN 16-059 Supplement 1 (Reference 1).

Modern Fuel:

The differences in the analysis basis for the ABWR standard design (as described in the DCD) and the basis for the recent analysis of the ABWR plant using the current ECCS-LOCA evaluation model are provided in Table 1 below. The recent analysis is identified in Tables 1-4 in this Enclosure as Current ABWR Project Work.

Table 1 – Differences in ABWR Standard Design and Current Analysis ABWR Design

Parameter	ABWR DCD	Current ABWR Project Work
Fuel	GE 7	GE 14
Rod configuration	8x8	10x10
Peak Linear Heat Generation Rate (LHGR)	44 kW/m	44.0 x 1.02 kW/m
Active fuel length	146 inches	150 inches
Top of Active Fuel (TAF)	904.95 cm	915.1 cm
Dome pressure	7.28 MPaA	7.27 MPaA
Residual Heat Removal/Low Pressure Flooder (RHR/LPFL) max. time from signal to pump at rated speed	29 sec	35 sec
	cm above TAF	cm above TAF
Level 2	243.4 (RCIC)	233.2
Level 1.5	98.7 (HPCF)	88.5 (RCIC & HPCF)
Level 1	15.3 (RHR & ADS)	5.1 (RHR & ADS)

The critical ECCS parameters are the same between the two plants (e.g. High Pressure Core Flooder (HPCF) flow rate) except for the one Residual Heat Removal (RHR) response time noted above. The additional time for the RHR/LPFL pump to maximum speed for the current plant is more conservative than the ABWR DCD time. The initiation of both RCIC & HPCF on Level 1.5 for the current project is also a conservatism.

The reactor pressure vessel water level in the computer models are referenced to the Reactor Pressure Vessel (RPV) zero. To maintain the analysis with the same values relative to the RPV

[4] Letter NRC to GEH, "NRC Audit of GE-Hitachi Nuclear Energy Americas Topical Report NEDO-33173, Supplement 4-A, *Implementation of PRIME Models and Data in Downstream Methods*", dated October 22, 2012. (ADAMS ML12277A401)

zero the increase in the active fuel length for the modern fuel results in a change of the analytic limits in relation to the top of active fuel.

The general consistency of the bases between the DCD analysis and the recent analysis presented here would affirm the conclusion that were an analysis of the ABWR DCD design to be re-performed using the current ECCS-LOCA evaluation model, such results of the current ABWR plant would be representative of projected results. This includes all items of the intervening 10 CFR 50.46 series of reported changes and errors.

The PCT results with current codes and modern fuel are still below the projected values based on the 10 CFR 50.46 adder (200°F/114°C) to the existing ABWR DCD results. The core will remain covered following LOCA events and the calculated peak cladding temperature remains low compared to the results for operating plant BWRs. Table 2 shows the comparisons.

Table 2 – Comparison of PCT Results for ABWR Standard Design and Current ABWR Design

	ABWR DCD	**ABWR DCD**	**Current ABWR Project Work**	**Current ABWR Project Work**	**ABWR DCD PCT With Adder[5]**
	Break size (cm^2)	**°C**	**Break size (cm^2)**	**°C**	**°C**
Appendix K evaluation					
Steamline Inside Containment	985	552	983	642	666
Feedwater Line	839	542	840	641	656
RHR Shutdown Cooling Suction Line	792	542	769	640	656
RHR/LPFL Injection Line	205	542	205	640	656
High Pressure Core Flooder	92	542	82	640	656
Bottom Head Drain Line	20.3	542	20	639	656
Steamline Outside Containment	3939	621	3933	639	735
Based on upper bound values					
Steamline Inside Containment	3939	619	3933	671	733

[5] The 200°F (114°C) Adder is contained in Letter GEH to 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.

As shown by Table 2, the assessment of a conservative Adder to account for the change in PCT as a result of 10 CFR 50.46 change and error reporting (as presented in MFN 16-059 Supplement 1), is bounding with respect to expected results assuming an explicit calculation given resolution of these reported items in the current ECCS-LOCA evaluation model.

The peak oxidation has also been determined for the current ABWR project with current methodology and modern fuel. Table 3 provides the comparison.

Table 3 – Comparison of Maximum Local Oxidation for ABWR Standard Design and Current ABWR Design

Maximum Local Oxidation (all %)	**ABWR DCD**	**Current ABWR Project Work**
Appx K evaluation		
Steamline Inside Containment	0.03	0.04
Feedwater Line	0.03	0.04
RHR Shutdown Cooling Suction Line	0.03	0.04
RHR/LPFL Injection Line	0.03	0.04
High Pressure Core Flooder	0.03	0.04
Bottom Head Drain Line	0.03	0.04
Steamline Outside Containment	0.03	0.04
Based on upper bound values		
Steamline Inside Containment	0.03	0.07

As shown, the maximum cladding oxidation percentage for the current ABWR project still maintains a significant margin to the allowable value of 17%.

The five acceptance criteria for the ABWR DCD ECCS, as specified in 10 CFR 50.46, are the following:

1. The calculated maximum peak cladding temperature (PCT) shall not exceed 1204°C (2200°F).
2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with the water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5. The calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core after any calculated successful initial operation of the ECCS.

The following is an update to the ABWR DCD FSER (NUREG-1503), Table 6.2, “Demonstration of Compliance with ECCS Criteria,” where GEH demonstrated compliance with the first three of these criteria as shown on Table 4 below:

Table 4 – Comparison Results for Three 10 CFR 50.46 Criteria

	Maximum		
	NRC FSER Table 6.2; From Break Analyses	**Current ABWR Project Work**	**Allowable**
Peak cladding temperature, °C (°F)	621 (1149)	671 (1240)	1204 (2200)
Maximum cladding oxidation, %	0.03	0.07	17
Maximum total hydrogen generation, %	0.03	0.07	1

For the ABWR, there is no core un-covery and, hence, a coolable core geometry is maintained. Long-term cooling is ensured by the use of redundant RHR systems that have adequate water sources available to keep the core covered and transfer the decay heat generated in the reactor core to the ultimate heat sink.

The ABWR ECCS meets 10 CFR Part 50, Appendix A, General Design Criterion 17, in that the safety function for each ECCS system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of the anticipated operational occurrences; and (2) the core will be cooled and containment integrity and other vital functions will be maintained in the event of postulated accidents.

<u>Conclusion of Comparison of ABWR Bounding PCT to Modern Analysis:</u>

Therefore, the current ABWR analysis with the modern methodologies and fuel is bounded by combination of the conservative Adder contained in Reference 1 and the existing ABWR DCD analysis results. The differences between the ABWR standard design and the current ABWR project are insignificant and do not impact the PCT results. The current ABWR ECCS analysis used for the comparisons is available for audit by the NRC staff, if desired.

Review of Additional Safety Analysis:

The impacts of the model changes described in MFN 16-059 Supplement 1 (Reference 1) on other ABWR safety analyses has been reviewed and are summarized in the following table.

Table 5 – Impacts of Model Changes on ABWR Safety Analyses

Item	Station Blackout (Appx 1C)	RPV Fluence (5.3)	Decay Heat (5.4)	Containment (6.2)	Combustible Gas (6.2)	LOCA (6.3)	Radiological (15)	Transients (15)	ATWS (Appx 15E)	PRA Success Criteria (19)
Original Analysis						Y				
1996-01		P			P	Y		P	P	Y
1992-02						Y				Y
2001-02						Y				Y
2001-04						Y				Y
2002-02						Y				Y
2002-03						Y				Y
2002-04						Y				Y
2003-01						Y				Y
2003-03						Y		P	P	Y
2006-01						Y				Y
2012-01						Y		P	P	Y
2014-01						Y				Y
2014-02						Y				Y
2014-03						Y				Y
2014-04						Y				Y

P- Potential impact with details provided below.
Y -Yes, impacted and details provided in MFN 16-059 Supplement 1.

The impacts of the model changes for each of the safety analyses are described below.

- Station Blackout – The analysis does not use the same models that are used for the ECCS-LOCA analysis, so there is no impact on this analysis.
- 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) is the only potential change to the RPV fluence analysis. Further review of the issue determined

that it was an incorrect active fuel rod number in the SAFER analysis only. Since the RPV fluence analysis does not use the SAFER methodology, it is unaffected.[6]

- 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. 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 – See MFN 16-059 Supplement 1 for details of impacts.
- 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 - The ABWR DCD transient analysis does not use the same models that are used for the modern ECCS-LOCA analysis except for TASC. A review of the model change effects has determined there are three items that need to be evaluated for transients.
  - The incorrect active fuel rod number (LOCA analysis item 1996-01) is the first potential change to the transient analysis and further review determined that the incorrect fuel rod number was in the SAFER analysis only. Therefore, there is no impact on transient analysis since SAFER is not used.
  - The second potential impact to the transient analysis is the steam separator pressure drop (LOCA analysis item 2003-03). If the transient analysis ABWR DCD code (ODYN) uses the same steam separator pressure drop as the LOCA analysis the impact on the results would be bounded by the conservative results of at least 30 psi in the predicted peak pressure.[7]
  - The last potential change is the PRIME Implementation (LOCA analysis Item 2012-01) with ODYN analysis. As described in NEDC-33173 Supplement 4-A, Revision 1, “Implementation of PRIME Models and Data in Downstream Methods”, future transient analysis will use PRIME as the source of inputs for single core average gap conductance value, with optional axial multipliers. In addition, the PRIME based thermal conductivity will be used. This same change will be applied to

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[6] Letter GEH to NRC, MFN-088-96, “Reporting of Changes and Errors in ECCS Evaluation Models”, dated June 28, 1996.
[7] NEDC-31336P-A, General Electric Setpoint Methodology, Section 4.8.1. This is based on NEDO-24154-A, Vol 2 “Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors”, Figures 3-9 to 3-14, August 1986.

TASC. These changes were approved in the Safety Evaluation associated with the LTR[8]. The impact will be on the Critical Power Ratio (CPR) calculations. As shown in ABWR DCD Table 15.0-2 there is significant margin to the MCPR limits since the limiting ΔCPR is 0.10 for the transient cases. Therefore, any change due to PRIME would not change the overall conclusion of the transient performance of ABWR.

The all reactor internal pump (RIP) trip event is the event in Chapter 15 that results in the determination of PCT. The ABWR DCD states that the resultant PCT is less than 600°C. This same event with modern fuel and methodologies results in the PCT of less than 500°C. Therefore, there is no change required to the ABWR DCD.

- ATWS – Same results as transients.
- 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 200°F (114°C) adder was added to the reported values in the DCD. There is no impact on the conclusion of the evaluation. The remaining PRA evaluations were performed using the MAAP code and it is independent of the ECCS-LOCA models.

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

In conclusion, there are multiple ABWR ECCS model changes that were summarized in the GEH Letter MFN 16-059, Supplement 1 (Reference 1). However, the only impacted safety analyses were the ECCS-LOCA and the PRA ECCS success criteria and the resultant bounding change on PCT was 200°F (114°C). Therefore, the ABWR DCD is modified for these impacts and no other DCD changes are necessary to address the new information provided in Reference 1 regarding the adjustments in the ECCS model since the original design certification.

Impact on DCD:

The DCD Tier 2 Tables 1.6-1 and 6.3-4 and Subsections 6.3.3.7.1, 6.3.7, 19.3.1.3.1 and 19.3.5 are being revised. The ABWR DCD R6 marked up pages are provided in Enclosure 2.

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[8] Letter to NRC to GEH, Final Safety Evaluation for GE Hitachi Nuclear Energy Americas Topical Report NEDO-33173, Supplement 4, “Implementation of PRIME Models and Data in Downstream Methods”, Sept 9, 2011. ML112440235