Enclosure 1

M170209

GEH Revised Response to RAI 06.03-3 and Responses to RAIs 06.03-4 through 06.03-9

NRC Request for Additional Information 06.03-3

In parts B.1, B.2, and B.3 of RAI 06.03-2, the staff requested that GE Hitachi Nuclear Energy (GEH) provide a description of the use of certain materials in the Advanced Boiling Water Reactor (ABWR) containment, how the design establishes limits on the quantities of those materials, and how the materials were evaluated for chemical effects. The staff requested this information in accordance with 10 CFR 52.59(a) (2014) as part of determining if the ABWR design complies with requirements in 10 CFR 50.46(b)(5)(1997) for emergency core cooling systems (ECCS). Chemical effects may increase strainer and reactor core clogging and form deposits on the fuel.

Follow-Up Question on the response to RAI 06.03-2, Parts B.1 and B.3

The staff requested the specific information in B.1 and B.3 of RAI 06.03-2 because the quantity of a material affects the amount of corrosion product it can generate in a given time period. Limiting the amount of materials contributing to corrosion products in the pool limits the amount of corrosion that can occur and the uncertainty in the chemical effects evaluation.

Based on the latest revision to GEH's RAI response to RAI 06.03-2 dated February 23, 2017 (ADAMS Accession number ML17055C593), it is not clear to the staff how chemical effects from all of the materials are addressed. Therefore, the staff requests the following information:

1. *The response to B.1.b indicates that a large surface area of zinc (galvanized steel) could be exposed to the post-loss-of-coolant accident (LOCA) water and corrode, depending on the pH, temperature, and location of the galvanized steel. The response to B.1.b also states that this zinc will not make a significant contribution to corrosion products in the suppression pool. Provide the basis for disregarding galvanized steel corrosion and the potential for corresponding zinc chemical effects. Address how the location of the galvanized steel (i.e., communication with the suppression pool), the corrosion rate predicted for the pH and temperature conditions, the solubility of zinc under the expected pool conditions, the amount and type of zinc precipitate formed in the pool, etc. were considered.*
2. *The response to B.1.d states that carbon steel is not a material of concern because it is not used in containment. DCD Section 6.1.1.1.2 indicates uncoated carbon and low-alloy steel is used in Engineered Safety Features components, and that corrosion is expected. If the iron in these components is released into the post-LOCA pool, describe how it was evaluated for chemical effects and the basis for concluding it would not be a concern.*
3. *The response to B.1.e.i refers to the steel liner plate that isolates concrete from the post-LOCA fluid. If any of the coated carbon steel liner is within the zone of influence for the coating, describe how the potential chemical effects from the exposed carbon steel were evaluated.*

Follow-up Question on the response to RAI 06.03-2, Part B.2

In B.2, the staff requested the ranges and timing of pH, pool temperature, and boron concentration following a LOCA. The response stated that:

1. *The pH is maintained in the range 5.3 – 8.6*
2. *No boron is present because the Standby Liquid Control System will not be used in a LOCA.*

The staff requested the pH and temperature ranges and transients because of the effect these parameters have on corrosion and precipitation, possibly leading to chemical effects at the strainer or in the reactor core. The staff requested the boron concentration range and transient because of the effect on pH and corrosion. The staff requested the transient behavior because corrosion and precipitation can depend on the sequence of conditions favoring corrosion and precipitation. The response described why boron would not be present but did not address transient behavior of pH or temperature. Therefore the staff requests the following information:

1. *Describe how the pH during the post-LOCA period is maintained in the same range as during operation. (The response to Question B.2 suggests that the Suppression Pool Cleanup System (SPCU System) maintains the pH in the range 5.3-8.6. According to DCD Section 9.5.9, the SPCU function is terminated during a LOCA.)*
2. *Describe the transient temperature and pH behavior during the post-LOCA period and how it was determined.*
3. *How do these predictions of pH account for the potential generation of strong acids (e.g., nitric and hydrochloric) from radiolysis, and subsequent reduction of pH in the pool if it is unbuffered and the SPCU System is not in use.*
4. *If a combined license (COL) applicant proposed using the SLC System during a LOCA, the application would need to evaluate the impact on chemical effects, ECCS strainer head loss, and downstream effects. Describe how this possibility will be addressed, for example by proposing a COL item requiring the applicant to perform this evaluation and submit the results for NRC review.*

GEH Revised Response to RAI 06.03-3

This is a revised response to RAI 06.03-3 to address certain information discussed in a public teleconference held on June 29, 2017. To the extent practical, the revisions to the response are shown in a track changes format.

This response addresses the questions above. Note that there are references in this response that are GEH proprietary internal documents that are available for NRC audit at GEH facilities. The NRC questions are repeated and underlined.

A. Follow-Up Question on the response to RAI 06.03-2, Parts B.1 and B.3

The staff requested the specific information in B.1 and B.3 of RAI 06.03-2 because the quantity of a material affects the amount of corrosion product it can generate in a given time period. Limiting the amount of materials contributing to corrosion products in the pool limits the amount of corrosion that can occur and the uncertainty in the chemical effects evaluation.

Based on the latest revision to GEH's RAI response to RAI 06.03-2 dated February 23, 2017 (ADAMS Accession number ML17055C593), it is not clear to the staff how chemical effects from all of the materials are addressed. Therefore, the staff requests the following information:

1. *The response to B.1.b indicates that a large surface area of zinc (galvanized steel) could be exposed to the post-loss-of-coolant accident (LOCA) water and corrode, depending on the pH, temperature, and location of the galvanized steel. The response to B.1.b also states that this*

zinc will not make a significant contribution to corrosion products in the suppression pool. Provide the basis for disregarding galvanized steel corrosion and the potential for corresponding zinc chemical effects. Address how the location of the galvanized steel (i.e., communication with the suppression pool), the corrosion rate predicted for the pH and temperature conditions, the solubility of zinc under the expected pool conditions, the amount and type of zinc precipitate formed in the pool, etc. were considered.

GEH Response: Exposed metallic zinc in the ABWR containment is limited to galvanized steel in ladders, ductwork, unistruts, cable trays, conduit and grating. Galvanized component material properties are specified by applicable ASTM material requirements. While galvanized components are not considered coatings under the RG 1.54, Coatings Program, the amount of zinc used inside the containment structure is considered (under the Service Level I, II and III Protective Coatings Applied to Nuclear Power Plants), an inventoried item for review and is, therefore, included under the COL Applicants Protective Coatings Program defined under DCD Section 6.1.3.1.

Galvanized steel structures and components are installed in the ABWR containment outside normally wetted areas. DCD Table 3I-2, Thermodynamic Environment Conditions Inside Primary Containment Vessel Plant Normal Operating Conditions, lists environmental conditions to which galvanized steel structures and components are exposed. General corrosion of these items in drywell conditions (moist nitrogen surroundings) could result in debris collecting in the suppression pool. Under normal plant operating conditions, the maximum suppression pool water and wetwell airspace temperature is 35°C or less. The general corrosion of galvanized steel structures and components is evaluated under latent debris terms in the ECCS suction strainer sizing (see DCD Section 6C.5 and NEDE-33878P).

Under design basis LOCAs inside primary containment, latent debris from galvanized steel structure and component corrosion deposits is transported to the suppression pool subject to ABWR suppression pool conditions (reactor water quality) with a pH of 5.3 to 8.~~6~~9, as listed in DCD section 3I.3.2.3. Note: The pH range of 5.3 to 8.6 was listed in error. DCD Table 3I-12, Thermodynamic Environment Conditions Inside Primary Containment Vessel Plant Accident Conditions, lists a maximum wetwell temperature of 122°C (251.6°F). The initial pool water temperature may rise to a maximum of 76.6°C (169.9°F) at 30 minutes. The continued release of decay heat after the initial blowdown may result in suppression pool temperatures as high as 88.7°C (191.7°F). These are the pH and temperature conditions that galvanized steel could be exposed under LOCA conditions. The Residual Heat Removal (RHR) System is available in the Suppression Pool Cooling mode to control the pool temperature. Heat is removed via the RHR heat exchanger(s) to the Reactor Building Cooling Water (RCW) System and finally to the Reactor Service Water (RSW) System.

BWROG Report, NWT 863, Review of Boiling Water Reactor Material Dissolution in Post- LOCA Containment Systems [ML 1 4328A639] includes test data (Figure 6-7) that represents zinc release from galvanized carbon steel to air saturated demineralized water at temperatures to 200°C. A conservative average Zn release rate of 0.05 gm/m^2-hr over a period of 100 hrs. was estimated for a pH range of 6 to 7. Using this corrosion rate, the weight of free zinc would be estimated for a 100- day period as follows:

Assume BWR containment= 1/1 0 vol of PWR containment with 1/1 0 of galvanized steel as listed in WCAP 1 6530-NP Table 3.1-1 with the maximum galvanized material to recirc water volume is 19.47 ft^2 / ft^3

assuming sp vol = 1 26,427 ft^3 [Ref DCD Tier 1 T2.14.1 (10)]

galvanized area = 126,427 ft^3 x 19.47 ft^2/ ft^3

galvanized area = 2,461,533 ft^2 = 228,676 m^2

use 1/1 0 of the PWR galvanized steel area = <u>22,868 m^2 [Engineering Estimate</u>

It was assumed un-submerged material did not contribute to zinc releases after termination of the spray phase as described in WCAP-16530. The maximum Zone of Influence (ZOI) determined for a non-domestic ABWR was 4530.75 ft^3 (128.3 m^3) during a MSL LOCA (reference GEH internal document 31113-0A51-2104 Section 6.1).

The galvanized area is adjusted to reflect that only a portion of the total galvanized inventory in the ABWR containment will be exposed to LOCA wetting conditions (ZOI):

A ratio of maximum ZOI volume (128.3 m^3) to total ABWR net free volume (5960 m^3 is total free volume ABWR drywell and wetwell ref: DCD Table 6.2-2) to was used to adjust the total galvanized steel surface subjected to LOCA:

Adjusted galvanized area = 22,868 m^2 (galvanized area) x 128.3 m^3 / 5960 m^3

Adjusted galvanized area = <u>493 m^2</u> = <u>5306 ft^2</u>

LOCA mission time 100 days x 24 hr/day = 2400 hr [Ref DCD Section 1A.2.31]

Minimum SP Volume = 3580 m^3 [Ref DCD Tier 1 T2.14.1 (10)]

Zinc release rate 0.05 gm/m^2-hr [Ref. BWROG Report, NWT 863, Review of Boiling Water Reactor Material Dissolution in Post- LOCA Containment Systems [ML 14328A639]

Zinc release= 0.05 gm/m^2-hr x 2400 hr x 493 m^2

Zinc release = 59,160 gm = <u>130 lbs</u>

Zinc SP concentration (ppm) = gm zinc /10^6 gm sp water 960 kg/m^3 (density at 207 C max temperature [Ref DCD Section 5.4.7.2.2])

Zinc SP concentration (ppm) = 59,160 gm Zinc x 1 x 10^6 I (960 kg/m^3 x 3580 m^3 x 1000 gm/kg)

Zinc SP concentration (ppm)= <u>17.2 ppm</u>

Due to the amount of exposed zinc (as galvanized steel) in BWR drywells, zinc-based precipitates may be a concern, such as zinc hydroxide (Zn(OH)2) which can form in slightly alkaline environments. The solubility of zinc ((Zn(OH)2)) has been estimated in an aqueous solution at varying pH and temperature conditions. It can be shown that zinc hydroxide will not precipitate out of solution in significant amounts at ABWR suppression pool conditions of pH and temperature.

NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations, March 2008 (ML080380214), notes that results provided in WCAP- 16530-NP, Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191, consistent with previous work such as the Integrated Chemical Effects Test (ICET) program, show that the predominant chemical precipitates are aluminum oxyhydroxide, sodium aluminum silicate and calcium phosphate (for plants using trisodium phosphate for pH control). Other minor silicate materials may also be generated (e.g., calcium aluminum silicate or zinc silicate), but the contribution of these materials is expected to be small relative to the predominant precipitates (i.e., less than 5 percent). On this basis, the chemical model considers only the release rates of aluminum, calcium and silicate. Other chemical species, such as zinc, may be ignored.

Updated GEH Response: This RAI response is updated to apply more appropriate BWR conditions and assumptions for the assessment of zinc (galvanized steel) deposition and precipitation in the ABWR suppression pool (SP). Several assessment parameters are revised from the previous response.

1. The total amount of zinc (galvanized steel) assumed in the ABWR containment is now developed from BWROG survey results and not from PWR data. The amount of zinc available for dissolution in the SP is estimated assuming using worst case (greatest inventory) BWR survey data [reference BWROG presentation provided Dec 2, 2015 during NRC public meeting; ML15335A419].
2. The estimation of the release rate of zinc under ABWR post-LOCA conditions was reviewed. An average release rate of 0.05 gm/m^2-hr was used for previous assessment applying test results from BWROG Report, NWT 863, Review of Boiling Water Reactor Material Dissolution in Post- LOCA Containment Systems Figure 6-5 Average Zinc Release Rate during Galvanized Coupon Exposure to Demineralized Water [ML14328A639 submitted by BWROG-14064; ML14328A636]. This release rate is conservative when compared to the zinc release rate equation provided in WCAP-16530-NP-A, Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191. The average release rate of 0.05 gm/m^2-hr is retained for this refined assessment.
3. The ABWR ECCS mission time under LOCA conditions was reviewed and is revised from 100 days to 30 days, consistent with NRC guidance, as described further below.
4. The zinc hydroxide estimated in the ABWR SP during post-LOCA conditions is compared to published industry data for the solubility of zinc in water to predict if the zinc in the SP would become a saturated solution, possibly plating out on the ECCS suction strainer surfaces. The most limiting SP temperature and pH are used to predict minimum zinc solubility (maximum precipitation).

The total amount of zinc (galvanized steel) assumed in the ABWR containment is **47,989 ft^2 total zinc**. This value of zinc is based on the BWR with the greatest zinc inventory surveyed. This is considered a bounding value with the average zinc inventory for all BWRs surveyed at 18,253 ft^2 total zinc [reference BWROG presentation provided Dec 2, 2015 during NRC public meeting; ML15335A419]. ABWR zinc inventory, limited to galvanized steel materials, will be less than the average legacy BWR due to the lower amount of zinc in containment.

The maximum Zone of Influence (ZOI) determined for a non-domestic ABWR was 4,530.75 ft^3 (128.3 m^3) during a MSL LOCA (reference GEH internal document 31113-0A51-2104 Section 6.1).

The galvanized area is adjusted to reflect that only a portion of the total galvanized inventory in the ABWR containment will be exposed to LOCA wetting conditions (ZOI), as follows:

A ratio of maximum ZOI volume 4,531 ft^3 (128.3 m^3) to ABWR non-wetted volume 259,563 + 210,475 = 470,038 ft^3 (ABWR drywell and pressure suppression pool free volume ref: DCD Table 1.3-3) was used to adjust the total galvanized steel surface subjected to LOCA:

$$4{,}531\ ft^3 / (259{,}563 + 210{,}475) = 4{,}531\ ft^3 / 470{,}038\ ft^3 = 0.00964$$

Adjusted galvanized area = 47,989 ft^2 (galvanized area) x 0.00964

Adjusted galvanized area = 463 ft^2 (**43 m^2**) (ABWR galvanized area subject to LOCA ZOI)

The average zinc release rate of **0.05 gm/m^2-hr** was applied to this assessment. [ref: BWROG Report, NWT 863, Review of Boiling Water Reactor Material Dissolution in Post-LOCA Containment Systems Figure 6-5 Average Zinc Release Rate during Galvanized Coupon Exposure to Demineralized Water; ML14328A639]. This is a conservative value when compared to the zinc release rate equation provided in WCAP-16530-NP-A, Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191, as illustrated below:

> The zinc release rate can be estimated using Equation 6-3 of WCAP-16530-NP-A, Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191
>
> $RR=10^{[A + B(pHa) + C(1000/T)+ D(pHa)(pHa)+E(pHa)(T)/1000]}$
>
> Where:
>
> RR = release rate in mg/(m^2 min)
>
> A = -15.10693334
>
> B = -3.670953896
>
> C = 7.303961651
>
> D = 0.103589245
>
> E = 5.485050709
>
> pHa = initial pH 8.9 corrected to 25°C=8.8 pHa
>
> T (°K) temperature 122 °F= 323.15°K
>
> $RR=10^{[-15.11-3.67(8.8) +7.30(1000/323)+ 0.10(8.8)\text{^}2+5.485(8.8)(323)/1000]}$

$RR = 10^{-1.186} = 0.065\ mg/(m^2\ min) \times gm/1000mg \times 60\ min/hr = \underline{0.0040\ gm/m^2\text{-}hr.}$

The ABWR ECCS mission time for post LOCA performance is revised from 100 days to 30 days (**720 hours**), consistent with NRC guidance. Specific to the ECCS chemical effects, NRC guidance in NUREG/CR-6988, "Final Report – Evaluation of Chemical Effects Phenomena in Post-LOCA Coolant," indicates that, although the regulations in 10 CFR 50.46(b)(5) require that long-term cooling be maintained indefinitely ("for an extended period of time"[1]), 30-days is typically considered to be an appropriate time period to demonstrate ECCS functionality and that, beyond this time, the decay heat loading is small, making alternative cooling possible should ECCS functionality be lost.

ABWR DCD Section 6.2.3.1, Item (3), applies a 100-day duration for operational capability associated with SSC credited for secondary containment function (but a 30-day duration for radiological analysis).

- ABWR DCD 15.6.5, Loss-of-Coolant Accident (Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary)—Inside Containment, analyzes a LOCA for 30 days. This is consistent with SRP 15.0.3 that supports a 30-day duration for all leakage paths and a 30-day mission time post-LOCA duration.
- The 30-day ECCS post-LOCA mission time has been applied for ECCS evaluation for the ABWR design. Specifically, the NRC has accepted this 30-day post LOCA duration for the South Texas Units 3 and 4, ABWR COLA [reference STP ABWR FSER Chapter 6 (ML120830102) and Chapter 9 (ML15021A327)].

Based on the refined assessment using an adjusted BWR zinc inventory and realistic ABWR ECCS mission time, the amount of zinc deposited in the ABWR SP can be estimated:

Zinc release = $0.05\ gm/m^2\text{-}hr \times 720\ hr \times 43\ m^2$

Zinc release = 1548 gm = 3.4 <u>lbm</u>

Zinc SP concentration (ppm) = gm zinc /10^6 gm SP water 960 kg/m^3 (density at 207°C max temperature [Ref DCD Section 5.4.7.2.2])

Zinc SP concentration (ppm) = 1548 gm Zinc x 1 x 10^6 / (960 kg/m^3 x 3580 m^3 x 1000 gm/kg)

Zinc SP concentration (ppm) = 0.45 ppm

The amount of zinc in solution can be compared to the zinc solubility estimated for the bounding suppression pool pH and temperature. Industry literature provides solubility determinations for molarity of dissolved zinc in water [ref: Zinc Hydroxide: Solubility Product and Hydroxy-complex Stability Constants from 12.5-75 C, Randy A Reichle, Keith G. McCurdy, and Loren G Hepler Dept of Chemistry, University of Lethbridge, Lethbridge, Alberta, T1K3M4, received June 30, 1975 [Reference http://www.nrcresearchpress.com/doi/pdf/10.1139/v75-556].

[1] 10 CFR 50.46(b)(5) *Long-term cooling*. "After any calculated successful initial operation of the ECCS, 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." Also, see (1) NRC Safety Evaluation for NEI 04-07 (ML043280007), and (2) NUREG/CR-7011 (ML101400088).

The lowest zinc solubility is shown to occur at the highest suppression pool pH at a temperature of 50°C within the range of pH from 5.3 to 8.9. The pH of 5.3 to 8.6 was used in earlier response. This pH described in DCD section 6.1.1.2 is applicable to the Suppression Pool Cleanup (SPCU) System. A maximum pH of 8.9 is more applicable to the suppression pool conditions during a LOCA as described in DCD section 3I.3.2.3. Under a zinc solubility of 0.94 x10^{-5} moles Zn/kg water (interpolated to 8.9 pH) is estimated using Table 1 in the paper referenced in the paragraph above:

S = 0.94 x10^{-5} moles Zn/kg water x 988 kg/m^3 x 3580 m^3= 33.2 moles Zn

33.2 Moles Zn x 65.38 gm/mole = 2173 gm Zn = 2.2 kg (4.8 lbm)

<u>Summary</u>

Based on the refined assessment using an adjusted BWR zinc inventory and the more appropriate ABWR ECCS mission time, the estimated amount of zinc deposited in the ABWR SP of 1.6 kg (3.4 lbm) is less than the estimated zinc solubility in the SP of 2.2 kg (4.8 lbm). Therefore, excessive precipitation of zinc in the ABWR SP post LOCA is not expected.

- This corresponds to BWROG test results showing no visual indication of precipitate formation for testing of galvanized coupons in contact with Nukon insulation continued for over a month. [reference BWROG presentation provided Dec 2, 2015 during NRC public meeting; ML15335A419].
- The zinc releases are relatively small and can be ignored in chemical effects precipitation modeling [WCAP-16530-NP-A, Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191, Section 6.2.2)].

<u>2. The response to B.1.d states that carbon steel is not a material of concern because it is not used in containment. DCD Section 6.1.1.1.2 indicates uncoated carbon and low-alloy steel is used in Engineered Safety Features components, and that corrosion is expected. If the iron in these components is released into the post-LOCA pool, describe how it was evaluated for chemical effects and the basis for concluding it would not be a concern.</u>

<u>GEH Response:</u> While the internal surface of the containment is lined with welded steel plate to form a leak tight barrier, stainless steel plate or clad is used on wetted surfaces of the suppression chamber. Type 304L stainless steel or clad carbon steel plate is used for the containment liner in the wetted areas of the suppression pool as protection against any potential pitting and corrosion on all wetted surfaces and at the water-to-air interface area.

ECCS materials of construction are listed in DCD Table 6.1-1, Engineered Safety Features Component Materials. A contributor to the latent debris in the suppression pool is normal corrosion of carbon steel piping and components in the ECCS systems. This corrosion is applied as the sludge/corrosion product ABWR ECCS strainer load. The ABWR sludge generation rate is less than the typical operating BWR, therefore the assumed ABWR sludge load of 200 lbm (100 lbm per year with a two-year operating cycle) is considered reasonable. Furthermore, there is a COL Item in Section 6.2.7.3 of the ABWR DCD which requires the applicant to establish a method for maintaining a level of cleanliness that supports this assumption.

The NRC requested additional information regarding the basis for applying a 200 lbm sludge load for ABWR during the public teleconference held on June 29, 2017.

The ABWR ECCS Strainer debris load of 90.7 kg (200 lbm) for Sludge / Corrosion Products is based on realistic assumptions and ABWR operating experience. There is margin applied to this ABWR debris load.

Section 3.2.4.3.2 of the Utility Resolution Guidance (URG) (NEDO-32686-A) describes a survey of operating BWRs that measured the rate of sludge generation. The data, collected from 12 plants with Mark I, II, and III containment designs, indicated a median sludge generation rate of 88 lbm per year. The URG recommends a value of 150 lbm per year to bound these results unless a lower plant-specific value can be justified.

The ABWR design features many improvements over the conventional BWRs that will help to minimize the generation of sludge. Specifically, the suppression pool is equipped with a stainless steel liner, and many interfacing systems utilize stainless steel pipe, which reduces the generation of carbon steel corrosion products. The ABWR suppression pool is enclosed in a concrete compartment and protected from the drywell environment, unlike some containment designs from the BWROG survey which are subject to dirt and debris falling through grating into the pool.

The above considerations suggest the ABWR sludge generation rate would be less than the typical operating BWR, therefore the assumed ABWR sludge load of 200 lbm (100 lbm per year with a two-year operating cycle) is considered reasonable. As noted in the NRC Safety Evaluation for the South Texas Units 3 and 4 ABWR COLA [ML120830102], the ABWR sludge load is conservative based on operational information from the Toshiba Electric Power Company (TEPCO) on quantities of material obtained from the SPCU systems at Kashiwazaki-Kariwa, Units 6 and 7, which are the oldest operating ABWRs.

Furthermore, there is a COL Item in Section 6.2.7.3 of the ABWR DCD which requires the applicant to establish a method for maintaining a level of cleanliness that supports this assumption. In addition, the COL applicant maintains a Protective Coatings Program, described in Section 6.1.3.1, which indicates the total amount of protective coatings and organic materials used inside the containment that do not meet the requirements of ANSI N101.2 and Regulatory Guide 1.54

While no credit is taken for the non-safety related Suppression Pool Cleanup (SPCU) System, it does function to remove particulates and dissolved impurities from the suppression pool and can provide indications of increased debris loading during operation through SP chemistry monitoring and trending of SPCU filter differential pressure. This will inform the COL applicant if additional SP cleaning / inspection should be scheduled.

As discussed in response to A.1 (above) for zinc, the release rates for iron from uncoated steel in the bench tests were on the same order as the zinc release rates, so iron can also be ignored in chemical effects precipitation modeling.

3. The response to B.1.e.i refers to the steel liner plate that isolates concrete from the post-LOCA fluid. If any of the coated carbon steel liner is within the zone of influence for the coating, describe how the potential chemical effects from the exposed carbon steel were evaluated.

GEH Response: It is assumed that all coatings, regardless of qualification, are assumed to fail within the LOCA jet Zone of Influence (ZOI). Where a LOCA jet directly impacts a coated surface, it is conservatively assumed the jet will strip off all the applied coating in the affected area without regard to coating qualification as discussed in NEDO-32686-A, Utility Resolution Guide for ECCS Suction Strainer Blockage. This will expose the carbon steel liner plate to the LOCA jet. Since this mechanism of debris generation is concentrated within the ZOI and of short duration, significant generation of corrosion or erosion products from the exposed steel liner are not expected.

Testing (described in NEI 04-07 Vol 2 NRC SER ML050550156_REVIEW OF NEI GUIDANCE APPENDICES Review of Appendix A, "Defining Coating Destruction Pressures and Coating Debris Sizes for DBA-Qualified and Acceptable Coatings in Pressurized Water Reactor (PWR) Containments) concluded that erosion was the primary mode of coating degradation from interaction with the waterjet in all test cases. The un-top-coated inorganic zinc coating failed at a distance up to 3 times greater than the epoxy. The industry concluded that a damage pressure of 333 psig for un-top-coated inorganic zinc and 1000 psig for epoxy systems should be used as the corresponding coating destruction pressures. Testing showed that an elevated surface temperature impacted the amount of coating degradation and increased fluid jet temperature resulted in coating degradation at lower jet pressures.

The rational that supports ignoring potential chemical effects precipitation modeling for iron (uncoated steel) was provided in response to A.2 above.

B. Follow-up Question on the response to RAI 06.03-2, Part B.2

In B.2, the staff requested the ranges and timing of pH, pool temperature, and boron concentration following a LOCA. The response stated that:

1. The pH is maintained in the range 5.3 – 8.6

2. No boron is present because the Standby Liquid Control System will not be used in a LOCA.

The staff requested the pH and temperature ranges and transients because of the effect these parameters have on corrosion and precipitation, possibly leading to chemical effects at the strainer or in the reactor core. The staff requested the boron concentration range and transient because of the effect on pH and corrosion. The staff requested the transient behavior because corrosion and precipitation can depend on the sequence of conditions favoring corrosion and precipitation. The response described why boron would not be present but did not address transient behavior of pH or temperature. Therefore the staff requests the following information:

1. Describe how the pH during the post-LOCA period is maintained in the same range as during operation. (The response to Question B.2 suggests that the Suppression Pool Cleanup System (SPCU System) maintains the pH in the range 5.3-8.6. According to DCD Section 9.5.9, the SPCU function is terminated during a LOCA.)

GEH Response: As described in DCD section 6.1.1.2, Demineralized water from the condensate storage tank or the suppression pool, with no additives, is employed in the core cooling water and containment sprays.

During operation, water in the 304L stainless steel-lined suppression pool is maintained at high purity (low corrosion attack) by the Suppression Pool Cleanup (SPCU) System. In the event of a LOCA, the SPCU function is automatically terminated to accomplish containment isolation. Therefore, this nonsafety-related system is not credited during LOCA / Post LOCA conditions.

Since the pH range (5.3 - 8.~~6~~9) is maintained, corrosive attack on the pool liner (304L SS) will be insignificant over the life of the plant.

Because of the methods described above (coolant storage provisions, insulation materials requirements, and the like), as well as the fact that the containment has no significant stored quantities of acidic or basic materials, the post-LOCA aqueous phase pH in all areas of containment will have a flat time history. In other words, the liquid coolant will remain at its design basis pH throughout the event.

2. Describe the transient temperature and pH behavior during the post-LOCA period and how it was determined.

GEH Response: Under normal plant operating conditions, the maximum suppression pool water and wetwell airspace temperature is 35°C(95°F) or less. Under blowdown conditions following an isolation event or LOCA, the initial pool water temperature may rise to a maximum of 76.6°C (169.9 °F) at 30 minutes. The continued release of decay heat after the initial blowdown may result in suppression pool temperatures as high as 88.7°C (191.7°F). The Residual Heat Removal (RHR) System is available in the Suppression Pool Cooling mode to control the pool temperature. Heat is removed via the RHR heat exchanger(s) to the Reactor Building Cooling Water (RCW) System and finally to the Reactor Service Water (RSW) System. The containment response, including suppression pool temperature and level is described in DCD section 6.2.1.1.3.3, Accident Response Analysis, with suppression pool temperature transient behavior shown on the following DCD figures:

- Figure 6.2-7, Temperature Response of the Primary Containment for Feedwater Line Break
- Figure 6.2-15, Temperature Time History for Long-term MSLB

Information in DCD Section 6.1.1.2, (coolant storage provisions and insulation materials requirements) and the fact that the containment has no significant stored quantities of acidic or basic materials, explain why the post-LOCA aqueous phase pH in all areas of containment will have a flat time history. The liquid coolant will remain at its design basis pH throughout the event.

3. How do these predictions of pH account for the potential generation of strong acids (e.g., nitric and hydrochloric) from radiolysis, and subsequent reduction of pH in the pool if it is unbuffered and the SPCU System is not in use.

Updated GEH Response: As discussed during the public teleconference held on June 29, 2017, GEH has revised this response for clarity.

GEH Response: The generation of chemical debris in the water chemistry that is representative of a BWR post-LOCA environment is expected to be less significant (as compared to PWRs) due to the typical BWR water chemistry and, more specifically, for the ABWR due to additional considerations in design features that minimize the potential for material interactions. Chemical debris generation through interaction with materials depends largely on the pH. Sources of acids and bases in a typical BWR include (1) cesium hydroxide produced by fission, (2) hydrochloric acid generated by the radiolysis of cable insulation, and (3) nitric acid generated by the radiolysis of water and air. These potential sources are not as significant in the ABWR due to design features as explained below.

In containments where no pH-control chemicals are present, the acidity or basicity of the water in the suppression pool will be determined by materials that are introduced into containment as a result of the accident itself. These materials may be fission products (i.e., cesium compounds), thermally produced products (i.e., core-concrete aerosols), or compounds produced by radiation (i.e., nitric acid). Sources of acids and bases in a typical BWR include (1) cesium hydroxide produced by fission, (2) hydrochloric acid generated by the radiolysis of cable insulation, and (3) nitric acid generated by the radiolysis of water and air. These potential sources are not as significant in the ABWR as explained below:

1) Cesium hydroxide (CsOH) is a fission product that could be transported to the ABWR suppression pool in post-LOCA conditions. CsOH is a strong base introduced into the primary containment and subsequently to the suppression pool with the release of cesium post-accident. Cesium should not be released significantly during a LOCA event without significant fuel damage. Therefore, it is not a source of increased pH in the ABWR SP following a LOCA.

2) Hydrochloric acid can be generated by the radiolysis of cable insulation [NUREG/CR-7172, "Knowledge Base Report on Emergency Core Cooling Sump Performance in Operating Light Water Reactors"]. Polyvinylchloride (PVC) cable jacket and insulation is the dominant source of the chloride ions via radiolysis. The chloride ions bond with hydrogen, creating acids. Cable jacket and insulation used in the ABWR is constructed of cross-linked polyolefin and cross-linked polyethylene, respectively, and does not have a major chloride component. [Reference WCAP-17938-NP, Rev. 2]. Polyvinyl chloride or neoprene cable insulation is not used in the ABWR [DCD Section 8.3.3.8.1].

3) Following a LOCA, the suppression pool pH could gradually trend downward due to postulated nitric acid formation in the reactor pressure vessel due to nitric acid generation from radiolysis. The effects of radiolysis in the core on primary chemistry pH are more pronounced in a PWR compared to a BWR due to the greater neutron fluence, average power density and coolant temperature.

   As noted in NUREG/CR 5950, "Iodine Evolution and pH Control," radiation dose rates ranged from -0.4 Mrad/h in a boiling water reactor (BWR) suppression pool to >5 Mrad/h in a pressurized-water reactor (PWR) sump. The effect of nitric acid produced by the irradiation of water and air is more significant in the PWR. ABWR DCD Section 19E.2.6.14, Suppression Pool pH Control, describes that without the production of cesium hydroxide (no fuel failure), the pH of the suppression pool will not drop to the acidic range within 24 hours of accident initiation. Therefore, nitric acid formation due to radiolysis will not have a significant impact on the source term.

Demineralized water, with no additives, is employed in ABWR core cooling water and containment sprays (see DCD Section 6.1.1.1.2 for a description of the water quality requirements). Leaching of chlorides from concrete and other substances is not significant. No detrimental effects occur on any of the ESF construction materials from allowable containment levels in the high-purity water. Thus, the materials are compatible with the post-LOCA environment.

Because of the methods described above (coolant storage provisions, insulation materials requirements, and the like), as well as the fact that the containment has no significant stored quantities of acidic or basic materials, the post-LOCA aqueous phase pH in all areas of the ABWR containment will have a flat time history. In other words, the liquid coolant will remain at its design basis pH throughout the event and strong acids will not be generated.

4. If a combined license (COL) applicant proposed using the SLC System during a LOCA, the application would need to evaluate the impact on chemical effects, ECCS strainer head loss, and downstream effects. Describe how this possibility will be addressed, for example by proposing a COL item requiring the applicant to perform this evaluation and submit the results for NRC review.

GEH Response: The ABWR Standby Liquid Control (SLC) System uses a dissolved solution of sodium pentaborate as the neutron-absorbing poison potentially added for reactivity control during accident mitigation. While the SLC system is capable of injecting borated water (pH control) for use as makeup water to the RPV in response to a LOCA, no credit is taken for buffering as a result of the sodium pentaborate injection into the RPV via the SLC system.

ABWR coolant storage provisions and insulation material requirements, as well as the fact that the containment has no significant stored quantities of acidic or basic materials, ensure that the post-LOCA aqueous phase pH in all areas of containment will have a flat time history (Ref: DCD section 6.1.1.2).

The pH of the suppression pool following a LOCA will be in the range of 5.3 to 8.~~6~~9, but the exact values as a function of time will depend on the actions taken by the operating crew in accordance with emergency procedures.

As discussed in Review of Boiling Water Reactor Material Dissolution in Post-LOCA Containment Solutions-NWT 863, November, 2013 [ML14328A639], Zinc release rate tests in borated water solutions at 8.5 pH (25°C) resulted in solution concentrations one to two orders of magnitude lower than those for demineralized water. Therefore, zinc release from galvanized carbon steel when exposed to borated solutions during a BWR LOCA event is not expected to be significant and was not given further consideration.

The licensing basis pH range of 5.3 to 8.~~6~~ 9 is specified in DCD Tier 2, Subsection 3I.3.2.3. Boron injection may be used to maintain the suppression pool pH within the specified range, but such use would be as directed by procedures. The COL applicant item for developing procedures is addressed in DCD Section 13.5.3. This includes development of plant operating procedures and emergency procedures, and specifically lists the SLC system and the conditions requiring use of the SLC system (see Sections 13.5.3.2 and 13.5.3.4).

In summary, the generation of chemical debris in the water chemistry that is representative of a BWR post-LOCA environment is expected to be less significant (as compared to PWRs) due to the typical BWR water chemistry and, more specifically, for the ABWR due to additional considerations in design features that minimize the potential for material interactions. ABWR coolant storage provisions and insulation material requirements, as well as the fact that the containment has no significant stored quantities of acidic or basic materials, ensure that the post-LOCA aqueous phase pH in all areas of containment will have a flat time history. Test results show zinc release is increased at lower pH. Also, solubility of zinc increases with increased water temperature resulting in zinc retained in solution at higher temperatures. At lower temperatures, zinc precipitate (and increased strainer headloss) is less of an issue since greater margin is available to ECCS pump NPSH. Test results show zinc release on exposure to borated solutions was minimal. Therefore, SLC operation and its impact on debris generation is minimal.

IMPACT ON DCD:

This response has no impact on the ABWR DCD.

NRC Request for Additional Information 06.03-4

General Design Criteria (GDC) 35, "Emergency Core Cooling," states, in part, that the emergency core cooling system (ECCS) safety function shall be to transfer heat from the reactor core following any loss of reactor coolant.

Appendix A, Tables A-4 through A-8 in NEDO-33878 (Public), Revision 1, dated May 2017, describe various ECCS flow paths that are credited during post-loss-of-coolant accident (LOCA) operation for the Advanced Boiling Water Reactor (ABWR) design. In the column titled Debris Ingestion Model, the applicant states that material will tend to settle out in low flow areas in piping. To support a finding under GDC-35, the applicant is requested to provide additional information to support its assessment that the settling of material in low flow areas in piping will not have adverse effects on system/component operation during the mission time. For example, the applicant is requested to address the quantity and type of material that will settle, locations where it will settle, and its impact on the performance of components in the applicable systems.

Revise the design control document (DCD) or NEDO-33878P as applicable.

GEH Response to RAI 06.03-4

The quantity and type of debris assumed downstream of the ECCS strainers is listed in Table A-2, ABWR Debris Source Term, of NEDE-33878P.

NEDE-33878P, Appendix A, Tables A-4 through A-8 provide the physical dimensions and properties of debris assumed downstream of the ECCS strainers. This information includes the settling velocity assumed for the specific type of debris evaluated.

NEDE-33878P Appendix A, Tables A-4 through A-8 assessed adverse effects on ECCS system/component operation for each applicable ECCS mode of operation during the ECCS mission time credited. The effect of settling of debris and the impact of this debris on ECCS system/component operation was reviewed for each ECCS by comparing the debris settling velocity to the system/component flow velocity during post LOCA operation. If the system/component flow velocity exceeds the debris settling velocity, it is assumed that minimal settling of debris will occur and performance of the ECCS components will not be adversely impacted. The flow through ECCS piping and components under design conditions exceeds the debris settling velocity with significant margin. Therefore, the impact of settling of debris in ECCS downstream the suction strainers will have minimal impact on system performance.

The ABWR ECCS mission time assumed for post LOCA performance of ECCS components is revised from 100 days to 30 days (720 hours), consistent with NRC guidance. Specific to the ECCS chemical effects, NRC guidance in NUREG/CR-6988, "Final Report – Evaluation of Chemical Effects Phenomena in Post-LOCA Coolant," indicates that, although the regulations in 10 CFR 50.46(b)(5) require that long-term cooling be maintained indefinitely ("for an extended period of time"), 30-days is typically considered to be an appropriate time period to demonstrate ECCS functionality and that, beyond this time, the decay heat loading is small, making alternative cooling possible should ECCS functionality be lost.

ABWR DCD Section 6.2.3.1, Item (3), applies a 100-day duration for operational capability associated with SSC credited for secondary containment function (but a 30-day duration for radiological analysis).

- ABWR DCD 15.6.5, Loss-of-Coolant Accident (Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary)—Inside Containment, analyzes a LOCA for 30 days. This is consistent with SRP 15.0.3 that supports a 30-day duration for all leakage paths and a 30-day mission time post-LOCA duration.
- The 30-day ECCS post-LOCA mission time has been applied for ECCS evaluation for the ABWR design. Specifically, the NRC has accepted this 30-day post LOCA duration for the South Texas Units 3 and 4, ABWR COLA [reference STP ABWR FSER Chapter 6 (ML120830102) and Chapter 9 (ML15021A327)].

The assessment assumed the ECCS design flowrate was maintained for the duration of the credited ECCS mission time. While reducing ECCS flow post LOCA may result in debris settling out in system low flow areas, adverse impact in ECCS performance is not anticipated based on the following:

- Lower ECCS flowrate will result in less debris entrained and transported to the ECCS suction strainer from the SP volume.
- Lower ECCS flowrate (through manual action) performed to adjust ECCS flow due to reduced heat load as LOCA recovery progresses provides additional margin on ECCS performance compared to design conditions. If degraded performance is indicated during this period of operation, flow could be increased to flush debris through the system on ECCS trains secured with design flow maintained in operating trains.

The impact of debris settling in instrument lines of ECCS system/components supporting post-LOCA functions was not detailed in the downstream assessment in NEDE-33878P. As described in response to RAI 06.03-5 below, ECCS instrument lines are installed above the horizontal plane of the process piping. The NRC staff safety evaluation for equivalent configurations concluded that there is no settling of debris in an instrument line installed above the horizontal plane. Therefore, no settling of debris in an instrument line in this orientation is expected.

NEDE-33878P and NEDO-33878 have been revised (Revision 2) to summarize debris settling impact on ECCS performance and reflect instrument line orientation that mitigates debris settling. These documents have also been revised to reflect credited ECCS post LOCA mission time to 30 days. RCIC is credited for 12 hrs. On this basis, NEDE-33878P demonstrates that the debris in the ECCS does not result in accumulation that would cause the system to become blocked or that the ECCS could not continue to perform its safety function during the post-LOCA conditions.

<u>NRC Request for Additional Information 06.03-5</u>

GDC 35 states, in part, that the ECCS safety function shall be to transfer heat from the reactor core following any loss of reactor coolant.

Appendix A, Tables A-4 through A-8 in NEDO-33878 (Public), Revision 1, dated May 2017, describe debris settling in instrument lines during post-LOCA operation for the ABWR design. In the column titled "Fluid Velocity through Component," the applicant states it is assumed that settling (instrument sensing lines/components) will occur when the flow velocity is less than the settling velocity for the debris type. The NRC staff evaluated debris settling in instrument lines as part of the review of the Pressurized Water Reactor Owner's Group topical report WCAP-16406-P, Revision 1, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191". Section 3.2.18, "Basis for Settling Velocity Multiplier for Bottom Mounted Instrument Lines" of the NRC staff safety evaluation for WCAP-16406P (Agencywide Documents Access and Management System Accession No. ML073520295) describes a settling velocity multiplier of seven to determine that entrained debris will not settle in bottom mounted instrument lines. A bottom mounted instrument line is defined as a line installed below the pipe horizontal plane. The NRC staff safety evaluation concludes that there is no settling of debris in an instrument line installed above the horizontal plane. To support a finding under GDC-35, the applicant is requested to provide additional information to describe whether any instrument lines are installed below the horizontal. If instrument lines are installed below the horizontal, the applicant is requested to describe the settling velocity multiplier used to determine that entrained debris will not settle in bottom mounted instrument lines.

Revise the DCD or NEDO-33878 as applicable.

<u>GEH Response to RAI 06.03-5</u>

The ABWR Project Design Manual for a non-US plant provides guidelines for locating process instrument connections (taps) on main process pipelines to ensure that fittings on the bottom of piping where they can collect crud are avoided. Therefore, ECCS instrument lines in service during post-LOCA operation are installed above the horizontal plane of the process piping. No settling of debris in an instrument line with this orientation is expected.

NEDE-33878P and NEDO-33878 have been revised (Revision 2) to reflect this ECCS instrument line configuration.

On this basis, NEDE-33878P demonstrates that the debris in the ECCS does not result in accumulation that would cause the system to become blocked or that the ECCS could not continue to perform its safety function during the post-LOCA conditions.

NRC Request for Additional Information 06.03-6

GDC 35 states, in part, that the ECCS safety function shall be to transfer heat from the reactor core following any loss of reactor coolant.

Appendix A, Tables A-4 through A-8 in in NEDO-33878 (Public), Revision 1, dated May 2017, describe the effect of post-LOCA debris on the residual heat removal (RHR), high-pressure core flooder (HPCF), and reactor core isolation coolant (RCIC) pumps during post-LOCA operation and concludes that the pumps will operate during post-LOCA conditions. For testing of safety related pumps, ABWR DCD, Revision 6, Section 3.9.6.1 specifies the design conditions under which pumps will be required to function. However, the ABWR DCD, Section 3.9.6.1 does not specifically address post-LOCA debris conditions under which pumps will be required to function. Therefore, the staff requests the applicant to address design and qualification requirements for the pumps during post-LOCA operation in Section 3.9.6.1 of the DCD. One acceptable method to demonstrate that a pump (including mechanical seal) can perform its specified function under all design basis conditions including post-LOCA debris conditions is ASME Standard QME-1-2007 as endorsed by RG 1.100, Rev. 3 "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants."

Revise the DCD as applicable.

GEH Response to RAI 06.03-6

NEDE-33878P, Appendix A, Tables A-4 through A-8 assess ECCS pump performance post LOCA under design debris loading for the required mission time. The assessment shows ECCS pumps will perform the required post LOCA function during design basis conditions, including design debris loading. This assessment used nuclear operating experience data for typical ECCS pumps and applied pump details for a specific non-US ABWR.

As described in NUREG /CR 2792, An Assessment of Residual Heat Removal and Containment Spray Performance Under Air and Debris Ingesting Conditions, ECCS pump performance degradation is expected to be negligible under LOCA conditions with generated debris at the pump. The data show that pump hydraulic performance degradation is negligible for particulate concentrations less than 1% by volume for a wide range of substances. Although data are limited, tests on mechanical wear of pumps indicate that the maximum calculated quantity of debris in the recirculating fluid is too small to impair pump operation as a result of material erosion.

The design debris loading downstream the ECCS suction strainers is estimated to be less than 1% by volume of the suppression pool water circulated by the ECCS pumps. Therefore, it is expected that the ECCS pumps will function post LOCA during the credited mission time.

ECCS pump performance for the specific plant as-built configuration will require demonstration of acceptable performance under design conditions including design debris loading. Demonstration of acceptable performance for as-built ECCS pumps is validated in accordance with QME-1 2007, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.

NEDE-33878P and NEDO-33878 have been revised (Revision 2) to reflect that ECCS pumps and associated mechanical seals will be qualified to operate with the post-LOCA fluids for at least 30 days (12 hours for RCIC pump), using qualification per QME-1 -2007 as endorsed by RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Revision 3, September 2009. These documents have also been revised to reflect credited ECCS post LOCA mission time to 30 days (RCIC is required to for 12 hrs).

On this basis, NEDE-33878P demonstrates that the debris in the ECCS does not result in accumulation that would cause the system to become blocked or that the ECCS could not continue to perform its safety function during the post-LOCA conditions.

ABWR DCD Revision 6 Tier 2 section 3.9.6.1 is revised to address post-LOCA debris conditions under which ECCS pumps are required to function. Tier 2 DCD Tables 1.8-20 and 1.8-21 are updated to reflect ECCS pump and component qualification per QME-1 2007 endorsed by RG 1.100 Revision 3.

DCD Revision 6 Tier 1 Tables 2.4.1, 2.4.2 and 2.4.4 are revised to add additional ITAAC requirements for as-built ECCS pump and component performance with design debris loading.

These DCD markups are provided in Enclosure 2 of this letter.

NRC Request for Additional Information 06.03-7

GDC 35 states, in part, that the ECCS safety function shall be to transfer heat from the reactor core following any loss of reactor coolant.

Appendix A, Tables A-4 through A8 in NEDO-33878 (Public), Revision 1, dated May 2017, describe the effect of post-LOCA debris on the operation of the RHR heat exchangers. In the column titled Debris Ingestion Model, the applicant states the following:

The RHR heat exchanger tube ID is 17.22 mm. The ECCS strainer will restrict debris to less than 3.18 mm. Therefore, the RHR heat exchanger will not become clogged from debris passing downstream of the ECCS suction strainer. However, the Auxiliary Equipment Evaluation column states that flow from the suppression pool is channeled through the shell side of the RHR heat exchangers and concludes that the heat exchangers will operate as designed during post-LOCA operation. To support a finding under GDC-35, the NRC staff requests the applicant to provide the following information and to specify vendor evaluation of the heat exchanger during the procurement process:

a. Describe the type, amount, and size of post-LOCA debris (if any) expected to pass through the RHR heat exchanger tubes.

b. For the shell side of the heat exchanger, the applicant is requested to specify that heat exchanger plugging, fouling, wear, and heat transfer performance during post- LOCA debris conditions (as specified in NEDO-33878) for the 100 day mission time is evaluated by the vendor during the procurement process and a certificate of compliance is provided to verify that the heat exchanger meets the design/procurement specifications.

Revise the DCD as applicable.

GEH Response to RAI 06.03-7

The statement associated with RHR flow through the shell side of the RHR HX [NEDO-32686 (URG) Vol 4]: is not applicable to the ABWR. As described in ABWR DCD section 5.4.7.1, the ABWR RHR heat exchanger has taken advantage of a design change that was made with respect to prior BWRs. ABWR has the reactor water flowing through the tube side of the heat exchanger, whereas, prior BWRs had the reactor water flowing through the shell side. The primary purpose for the change was to reduce radiation buildup in the heat exchanger by providing a more open geometry flow path through the center of the tubes, as opposed to the shell side construction of spacers, baffles, and low flow velocity locations, which can provide places for radioactive sludge to accumulate.

a) The quantity and type of debris assumed downstream of the ECCS strainers is listed in Table A-2: of NEDE-33878P, ABWR Debris Source Term.

   NEDE-33878P, Appendix A, Tables A-4 through A-8 provide the physical dimensions and properties of debris assumed downstream of the ECCS strainers. This information includes the settling velocity assumed for the specific type of debris evaluated.

b) Since the debris assumed downstream of the ECCS strainer does not contact the shell of the RHR heat exchanger during a postulated LOCA, heat exchanger shell side plugging, fouling, wear, and degradation of heat transfer performance is not applicable.

However, degradation of the tube side of the RHR heat exchanger was evaluated as described in NEDE-33878P and NEDO-33878.

- The RHR HX tube diameter exceeds the largest dimension of debris downstream of the RHR suction strainer. Therefore, plugging of the tubes is not considered credible.
- The debris velocity assumed through the RHR HX exceeds the settling velocity of the specific type of debris. Therefore, fouling of the tubes is not considered credible.
- The specified materials of the RHR heat exchanger subjected to debris (e.g. tubes-SA213-316/316L) will be a hard material that will resist wear and abrasion during the ECCS post LOCA mission time (revised) from 100 days to 30 days for RHR. Therefore, wear of the tubes is not considered credible.
- The RHR hear exchanger tube side flow geometry and orientation minimizes buildup of debris during operation. Debris will not settle out in the RHR heat exchanger. Therefore, degradation of the RHR heat exchanger heat transfer performance during the ECCS post LOCA mission time is not considered credible.

The RHR HX specifications require the vendor to meet performance requirements under design debris loading conditions. This will be validated through the procurement process with a certificate of compliance provided.

NEDE-33878P and NEDO-33878 have been revised (Revision 2) to clarify the flow path through the RHR heat exchanger and revise the credited ECCS post LOCA mission time to 30 days.

On this basis, NEDE-33878P demonstrates that the debris in the ECCS does not result in accumulation that would cause the system to become blocked or that the ECCS could not continue to perform its safety function during the post-LOCA conditions.

NRC Request for Additional Information 06.03-8

GDC 35 states, in part, that the ECCS safety function shall be to transfer heat from the reactor core following any loss of reactor coolant.

Appendix A, Tables A-4 through A8 in NEDO-33878 (Public), Revision 1, dated May 2017, describe the effect of post-LOCA debris on components and the applicant states that system wear for the mission time of 100 day is insignificant. To support a finding under GDC-35, the applicant is requested to provide the basis for stating that component wear for 100 days is insignificant. For example, describe wear rate evaluations performed for the individual components during the 100 days mission time and the determination that the final system flow rates are acceptable for post-LOCA operation. Revise the DCD or NEDO-33878 as applicable.

GEH Response to RAI 06.03-8

The ABWR ECCS mission time under LOCA conditions was reviewed and is revised from 100 days to 30 days, consistent with NRC guidance. RCIC is required for 12 hrs. Utilizing a more realistic mission time allows crediting typical ECCS component operating experience under similar post LOCA conditions.

As described in NEDE-33878P and NEDO-33878, the concentration of suspended solids in the SP water is estimated at 5130 ppm by weight [0.07% vol.] for non-fiber debris and 6.8 ppm by weight (0.018% vol.) fiber debris assuming the minimum SP volume and worst case debris volume.

Experimental data on the effects of particulates on pump hydraulic performance applied to ECCS type pumps show that pump performance degradation is negligible for particulate concentrations less than 1% by volume. [Ref: NUREG/CR 2792]. NUREG/CR 2792 notes conservative estimates of the nature and quantities of debris show that fine abrasives may be present in concentrations of about 0.1% by volume (about 400 ppm by weight). and that very conservative estimates of fibrous material yield concentrations of less than 1% by volume. Published data on the effects of particulates on pumps generally deal with particulate concentrations at many times these values.

The expected wear of non-rotating ECCS components such as piping, valves, heat exchangers, spargers and instrumentation during the 30 day post-LOCA mission time under design basis debris loading is also not expected to adversely impact the ECCS performance.

The ECCS post LOCA downstream effects assessment was performed using typical ECCS components. To ensure the as-built ECCS post LOCA performance is met, Core Cooling System Inspection, Test, Analysis and Acceptance Criteria (ITAAC) will be revised as indicated on the enclosed markup.

NEDE-33878P and NEDO-33878 have been revised (Revision 2) to update the credited ECCS post LOCA mission time to 30 days. RCIC is required to for 12 hrs.

On this basis, NEDE-33878P demonstrates that the debris in the ECCS does not result in accumulation that would cause the system to become degraded due to wear or that the ECCS could not continue to perform its safety function during the post-LOCA conditions.

NRC Request for Additional Information 06.03-9

GDC 35 states, in part, that the ECCS safety function shall be to transfer heat from the reactor core following any loss of reactor coolant.

RG 1.82 revision 4 states downstream blockage is a concern for tight-clearance valves. The applicant does not address blockage due to valves that are not in the fully open position. To support a finding under GDC-35, the applicant is requested to provide the basis for determining that blockage is not a concern for tight-clearance valves (such as throttle and check valves) that are not in the fully open position during post-LOCA operation. Revise the DCD or NEDO-33878P as applicable.

GEH Response to RAI 06.03-9

As described in NEDE-33878P and NEDO-33878, ECCS components in the flow path in service during post-LOCA modes of operation were evaluated from failure due to blockage under design debris loading. Tight clearance valves such as throttle and check valves were reviewed under this evaluation.

RHR

The RHR system has no tight clearance valves throttled during post LOCA operation that would be susceptible to blockage or binding. All RHR valves in the post LOCA lineup will be closed (i.e. isolate CST suction flow path) or fully open. As reflected on Table 1, Valve Position Chart, on Figure 5.4-11, Residual Heat Removal System PFD (Sheet 2 of 2), no RHR valves are throttled during post LOCA modes of operation. RHR minimum flow is maintained by a piping orifice rather than throttling of the minimum flow valve.

RHR system check valves installed in the main RHR pump discharge line, minimum flow line and jockey pump discharge line have active safety functions to open. These RHR valves are not susceptible to clogging, settling or wear. The clearances of these check valves prevent debris from adversely impacting the function of these components. The check valve material is carbon steel. Erosion or wear during the post LOCA credited 30-day mission time will not impact system performance.

RHR system orifice plates and SP and drywell spargers installed in the RHR process piping have safety functions to maintain flow. These RHR components are not susceptible to clogging, settling or wear. The clearances of these components prevent debris from adversely impacting the function of these components. The orifice and sparger material is stainless steel. Erosion or wear during the post LOCA credited 30-day mission time will not impact system performance.

HPCF

The HPCF system has no tight clearance valves throttled during post LOCA operation that would be susceptible to blockage or binding. All HPCF valves in the post LOCA lineup will be closed (i.e. isolate CST suction flow path) or fully open. As reflected on Table 1, Valve Position Chart, on Figure 6.3-1 High Pressure Core Flooder System PFD (Sheet 2 of 2), no HPCF valves are throttled during this mode of operation. HPCF minimum flow is maintained by a piping orifice rather than throttling of the minimum flow valve.

HPCF system check valves installed in the main HPCF pump suction, discharge and minimum flow line have active safety functions to open. These HPCF valves are not susceptible to clogging, settling or wear. The clearances of these check valves prevent debris from adversely impacting the function of these components. The check valve material is carbon steel. Erosion or wear during the post LOCA credited 30-day mission time will not impact system performance.

HPCF system orifice plates and reactor vessel sparger installed in the HPCF process piping have safety functions to maintain flow. These HPCF components are not susceptible to clogging, settling or wear. The clearances of these components prevent debris from adversely impacting the function of these components. The orifice and sparger material is stainless steel. Erosion or wear during the post LOCA credited 30-day mission time will not impact system

<u>RCIC</u>

The RCIC system has no tight clearance valves throttled during post LOCA operation that would be susceptible to blockage or binding. All RCIC valves in the post LOCA lineup will be closed (i.e. isolate CST suction flow path) or fully open. As reflected on Table 1, Valve Position Chart, on DCD Figure 5.4-9, Reactor Core Isolation Cooling System PFD (Sheet 2 of 2), no RCIC valves are throttled during this mode of operation. RCIC minimum flow is maintained by a piping orifice rather than throttling of the minimum flow valve. RCIC flow is varied by RCIC turbine speed by positioning the steam governor value to maintain system flow rather than throttling RCIC process valves. RCIC is required to support post LOCA function for 12 hrs.

The RCIC system check valve installed in the main RCIC pump discharge line has an active safety function to open. This RCIC valve is not susceptible to clogging, settling or wear. The clearances of this check valve prevent debris from adversely impacting the function of these components. The check valve material is carbon steel. Erosion or wear during the post LOCA credited 12-hr mission time will not impact system performance.

The RCIC system orifice plate and feedwater sparger installed in the RCIC process piping have safety functions to maintain flow. These RCIC components are not susceptible to clogging, settling or wear. The clearances of these components prevent debris from adversely impacting the function of these components. The orifice and sparger material is stainless steel. Erosion or wear during the post LOCA credited 12-hr mission time will not impact system.

NEDE-33878P and NEDO-33878 have been updated under revision 2 to reflect that ECCS pumps and associated mechanical seals will be qualified to operate with the post-LOCA fluids for at least 30 days (12 hours for RCIC pump), using qualification per QME-1 -2007.

as endorsed by RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Revision 3, September 2009. These documents have also been revised to reflect credited ECCS post LOCA mission time to 30 days (RCIC is required for 12 hrs).

On this basis, NEDE-33878P demonstrates that the debris in the ECCS does not result in accumulation that would cause the system to become blocked or that the ECCS could not continue to perform its safety function during the post-LOCA conditions.

ABWR DCD Revision 6 Tier 2 section 3.9.6.1 is revised to address post-LOCA debris conditions under which ECCS pumps are required to function. Tier 2 DCD Tables 1.8-20 and 1.8-21 are updated to reflect ECCS pump and component qualification per QME-1 2007 endorsed by RG 1.100 Revision 3.

DCD Revision 6 Tier 1 Tables 2.4.1, 2.4.2 and 2.4.4 are revised to add additional ITAAC requirements for as-built ECCS pump and component performance under design debris loading with the acceptance criteria per QME-1 applied.

Impact on DCD:

DCD Section	Change
Tier 1:	
• Table 2.4.1, Residual Heat Removal System o Design Commitment 4 c RHR Pump NPSH	Update the Inspections, Test and Analysis and Acceptance Criteria to reflect RHR pump as-built performance will consider design debris loading.
• Table 2.4.4, High Pressure Core Flooder System o Design Commitment 3 g HPCF Pump NPSH	Update the Inspections, Test and Analysis and Acceptance Criteria to reflect HPCF pump as-built performance will consider design debris loading.
• Table 2.4.4, Reactor Core Isolation Cooling System o Design Commitment 3 j RCIC Pump NPSH	Update the Inspections, Test and Analysis and Acceptance Criteria to reflect RCIC pump as-built performance will consider design debris loading.
Tier 2:	
• Table 1.6-1 Referenced Reports	NEDO-33878 and NEDE-33878P, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability, revision level updated to Revision 2 to incorporate changes under RAI 06.03-4 through RAI 06.03-9].

DCD Section	Change
• Table 1.8-20 NRC Regulatory Guides Applicable to ABWR	Add RG 1.100 Rev 3 to regulatory guides applicable to ABWR adding comment (2) scope that this revision is applicable for qualification of ECCS pumps per QME-1. Qualification of Active Mechanical Equipment Used in Nuclear Power Plants 2007 described in section 3.9.
• Table 1.8-21, Industrial Codes and Standards Applicable to ABWR	Add QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants 2007 to list of ASME standards applicable to ABWR referencing note (10) to limit scope to qualification described in DCD subsection 3.9.6.1.
• Section 3.9.6.1	The testing of safety related pumps is revised to address post-LOCA debris conditions under which ECCS pumps are required to function pumps with performance qualified per QME-1. Qualification of Active Mechanical Equipment Used in Nuclear Power Plants 2007.
• Appendix 6C Containment Debris Protection for ECCS Strainers	Updated 6C.3.3, Downstream Effects, to define aspects of downstream assessment documented in NEDE-33878P and update ECCS pump and component performance qualification per QME-1. Updated 6C.7. References, to reflect revision of NEDE-33878P and NEDO-33878.