Discussion Points for Public Meeting with NextEra Energy Point Beach, LLC, Related to Point Beach Containment Dome Truss License Amendment Request Review

Enclosure 1 to Response to Request for Additional Information (ADAMS Accession No. ML18102B164)

1. The proposed criteria for the containment structure concrete behind the liner in the response to RAI-1.a includes permitting localized concrete strain exceedance provided "[l]ocalized exceedance of strain limits does not significantly reduce containment structure shell strength." The licensee has not defined or explained the term "significantly reduce" in the proposed criteria.

Enclosure 2 to Response to Request for Additional Information (ADAMS Accession No. ML18102B164)

2. Page 28 of 163: The modified fragility for the containment isolation system (CIS) is not used in Table 9. The fragility is higher resulting in a lower failure probability. It is unclear to the staff why the modified fragility was not used and whether the change in risk results following convolution using the modified fragility will be higher than the ones reported.

3. Section 2.1.4.2 of the revised risk evaluation, Page 22 of 163: Based on the discussion, it is unclear to the staff how the selected approach captures the frequency of feed and bleed (F&B) successes and therefore, supports the exclusion of F&B as an initiator.

4. Section 2.3.2.3 of the revised risk evaluation, Page 30 of 163: The feedwater line break inside containment (FLBIC) initiating event frequency continues to use information from the Electric Power Research Institute (EPRI) report 302000079, Revision 3, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments." The steam line break inside containment initiating event frequency in Section 2.3.2.1 is derived from "SPAR [Standardized Plant Analysis Risk] Event Data and Results 2015 Parameter Estimation Update." The same source also contains the initiating event frequency for FLBIC. The basis for the continued use of the information from the EPRI report is unclear to the staff. Further, it is unclear whether using a different initiating event frequency for FLBIC will impact the reported results.

5. Section 5.3 of the revised risk evaluation, Page 42 of 163; Page 104 of 163: The demonstrably conservative thermal analysis includes F&B being questioned for all thermal initiators considered in the analysis. The inclusion of F&B and the resulting credit appears unjustified to the staff because of the following:

a. F&B is expected to be effective primarily for small breaks because the Reactor Coolant System (RCS) conditions during a medium and large break will make the bleed portion unnecessary. The initiating events are not separated for use in the demonstrably conservative thermal analysis. Therefore, it appears that F&B is being credited for initiators where it will be unnecessary and/or ineffective.

b. According to the results of the thermal-hydraulic calculations shown on page 34 of 163 of the revised risk evaluation, the temperature peak is reached at around 100 seconds. It appears to the staff that it would be unlikely to perform F&B within that timeframe. It is recognized that the analysis on page 34 of 163 is for Main Steam Line Break (MSLB). However, separate analyses have not been presented for cases where F&B would be effective to determine time available for F&B compared to the time for temperature rise in the truss.

6. Page 87 of 163 of the revised risk evaluation: The following items are unclear to the staff based on the information presented:

    a. The source of the failure probability data in the feedwater isolation valve fault tree,

    b. Justification for not modeling common cause failures, which do not appear to be modeled for the feedwater isolation valve fault tree. Further, the presence or absence of common cause failure modeling for other fault trees cannot be ascertained by the staff because those fault trees are not presented, and

    c. Whether gates other than the feedwater isolation and the safeguards actuation used inputs already in the licensee's National Fire Protection Agency Standard 805 (NFPA 805) model or new inputs were used (as well as the source of the data for those inputs).

7. Page 89 of 163 of the revised risk evaluation: The fault tree for Case 4 uses 1 of 2 containment spray failure gate instead of 2 of 2 as the case description in Table 10 (on page 90 of 163) states. It is unclear to the staff whether this is an inconsistency in the modeling and if so, what the impact of any changes is on the reported results.

8. Section F.3, Page 112 of 163 of the revised risk evaluation: The table provided for Section F.3 shows that pretty much all of the fragilities except the CIS were from the Individual Plant Examination of External Events (IPEEE). The description for the HHI [High Head Injection] top event states that a non-IPEEE fragility was used for the HHI seismic failure. Section 9.1.2.3 has a sensitivity for

fragility values where the "original IPEEE values were applied wherever updated or generic fragilities were applied…" It is unclear to the staff if the CIS and HHI fault trees are the only instances of change for the sensitivity in Section 9.1.2.3 or there were other unidentified instances.

9. Section F.5 of the revised risk evaluation, Page 119 of 163: The top event description for containment truss induced very small loss-of-coolant accident (LOCA) ("LOCA – CT induced very small LOCA") discusses a very small LOCA due to seal table failure from a falling truss member (debris). However, the cutsets shown from page 132 of 163 onwards have a small LOCA event with the probabilities corresponding to the seismically induced small LOCA. Further, the description for "LOCA – CT induced very small LOCA" top event states that a very small LOCA is modeled which requires Pilot Operated Relief Valves (PORVs) but in the event tree on pages 117 and 118 of 163, PORVs are not questioned and only HHI is questioned.

    a. The logic model for the "LOCA – CT induced very small LOCA" is unclear to the staff and appears to be inconsistent with the description.

    b. The staff is unable to determine the impact of considering only the very small LOCA due to seal table failure on the reported results for the demonstrably conservative case.

10. Section F.6 of the revised risk evaluation, Page 122 of 163: "Seismic-03" sequence has a core damage frequency (CDF) of 2.72E-7 from "FTREX." However, the first two cutsets on page 132 of 163, which are for the "Seismic-03" sequence, result in a value of 4.3E-7 which appears to the staff to be inconsistent with the value reported on Page 122.

11. Section F.7 of the revised risk evaluation, Page 129 of 163; The following items are unclear to the staff based on the information presented:

    a. The source of the random failure probabilities in the fault trees for the non-seismic failures of various top events in the CDF calculation (e.g. auxiliary feed water).

    b. The source of the random failure probabilities for the non-seismic failures of the containment isolation and the fan coolers.

    c. The modeling and coupling of the top events included from the IPEEE in the CDF event and/or fault trees as well as the containment isolation and fan coolers in the seismic Large Early Release Frequency (LERF) event and/or fault trees.

d. The failure probability used for the containment penetrations above 66 feet elevation. Section 5.2 of Enclosure 2 the submittal states that a failure probability of 0.1 was used for penetrations below the 66 feet elevation but does not clarify the value that was used for penetrations above 66 feet. Page 112 of 163 of Enclosure 2 describes the LERF seismic CT model and states that the containment isolation and containment fragilities are included but does not address how the 0.1 and/or 0.5 failure probabilities are applied.

e. The approach used to determine the change in LERF reported for the disposition on item 11 in the Table in Section 9.3 by including steam generator tube rupture events into LERF calculations.

12. Attachment G of the revised risk evaluation:

    a. Based on the fault trees presented in Attachment G it is unclear to the staff how assumption #5 on page 147 of 163 is incorporated.

    b. Based on the fault trees presented in Section G.4 and G.5, it appears to the staff that assumption #2 on page 147 of 163 is not being applied consistently.

    c. The approach used by the licensee for determining the “conservative” and “bounding” values for the failure probability of the structures, systems, and components in Attachment G (e.g., MSL “A” Break in Section G.2) in the context of the corresponding fault trees is unclear to the staff.

Enclosure 1 to Document Transmittal (ADAMS Accession No. ML18102B173):

13. Section 3.5.4: It is unclear to the staff how the main steam vent line failure probability of 0.9 is consistent with the assessment in Attachment F of Enclosure 2. Attachment G of Enclosure 2 includes failure probability of the main steam lines which is different from the main steam vent lines. Item 4.ii in Section 2.1.3 of Enclosure 2 provides a qualitative description of the main steam vent lines which appears to reach a different conclusion from Section 3.5.4 of the target assessment report.

14. Recognizing that the target assessment is semi-quantitative, the staff seeks to better understand the aspects of the assessment that provide assurance that the lack of verification or validation of certain assumptions used in target assessment will not impact the conclusions.