Point Beach Subsequent License Renewal Application Review August 5, 2021, Public Meeting Discussion Questions Proposed Aging Management of Irradiated Concrete and Steel Reactor Vessel Supports

The following discussion questions relate to the applicant's responses to the listed RAIs:

RAI 3.5.2.2.2.6-1

The RAI response indicates the estimate of the uncertainty was established using an existing reactor pressure vessel extended beltline uncertainty analysis. This statement implies the analysis may have been submitted to NRC in a prior review. If so, what is the reference for the analysis?

If not, clarify the types of parameters identified as having a potentially significant contribution to the core neutron source, reactor geometry, coolant temperatures, discretization, and modeling approximation uncertainties (i.e., provide some examples of the parameters for each of the mentioned 'categories'). Also, clarify whether the level of detail within the RPV model is commensurate with the PBN plant-specific model.

Additionally, is the estimated analytical uncertainty for the E > 1.0 MeV neutron fluence expected to be representative of the analytical uncertainty for the E > 0.1 MeV neutron fluence?

RAI 3.5.2.2.2.6-2

- SLRA Table 2.4-1, "Containment Structure and Internal Structural Components Subject to Aging Management Review," on page 2.4-6 (Attachment 2, page 5/13) states:

Component Type	Intended Function(s)
Primary shield wall (and biological shield wall)	Radiation shielding
	Shelter protection
	Structural support

  This entry contrasts statements made in the SLRA (see page 3.5-33) and promulgated during audit that the BSW has no structural function. Please clarify the BSW intended function(s).

- SLRA Table 3.5-1 (Attachment 2, page 8/13) is augmented by line item 3.1-1, 134. Confirm/clarify location of the inserted line item.

- In regard to the RMI insulation, the RAI claims (Attachment 2, page 3/13) that "the insulation specifications have been uploaded to the ePortal." Clarify the location of the uploaded specs in the ePortal."

  Attachment 2, page 3/13 claims that procedures are to be revised so that RMI insulation inspections will provide an assessment of its condition inside the PSW penetrations. A search of the ePortal does not reveal (a) revised procedure(s) in a DRAFT state incorporating "visual inspections for loss of

material (every refueling outage) and cracking (every ten years) on accessible portions of the RC piping RMI insulation and insulation jacketing." The RAI response also does not include a revision to SLRA Commitment 27 regarding revision of said procedure(s). If not necessary to include such, identify which of the listed bullets in Commitment 27 addresses visual inspections of the RMI insulation.

RAI 3.5.2.2.2.6-3

Revised SLRA Table 2.4-1 on page 2.4-5 (Attachment 3, page 4/10) indicates that the PSW's liner function is no longer associated with "Radiation Shielding." It's new safety function is "Structural Support." It is not clear what the structural support of the PSW steel liner is.

Revised SLRA Section 3.5.2.1.1 (Attachment 3, pages 8/10, 9/10, 10/10) indicates that the BSW liner will be monitored by Structures Monitoring AMP for distortion as a leading indicator of the RIVE effect. Table 3.5.2-1 (Attachment 3, page 7/10) lists the PSW steel liner to be monitored for distortion. There is no entry for the BSW steel liner being monitored for distortion during the SPEO. It is not clear how monitoring the PSW steel liner will reveal effects of RIVE for the BSW underlying concrete.

RAI 3.5.2.2.2.6-4

Based on reported fluence uncertainty of 20% (without the analytical uncertainty raised RAI 3.5.2.2.2.6-1 clarification) in the irradiated concrete damage threshold in the BSW concrete would exceed the clear cover of temperature steel. It is not clear whether bond effectiveness of temperature steel at selected BSW areas and concrete depths would be marginalized.

RAI 3.5.2.2.2.7-2 and RAI 3.5.2.2.2.7-4

RAI 3.5.2.2.2.7-2 states that "[t]he stresses were conservatively calculated based on the maximum loads regardless of the stress location being evaluated."

RAI 3.5.2.2.2.7-4 states that "the IRs are maximized at considered welded joint regions."

It is not clear whether the maximum IRs would occur in areas of high stress concentration, potentially leading to yielding and fracture.