Response to

Request for Additional Information No. 22, Revision 0

7/3/2008
U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19
SPLB Branch

Question 19-146:

In Section 19.2.4.2.1.2, on page 19.1-82, the FSAR states that in vessel hydrogen production was assessed to range between 48% (1,580 pounds of H2) and 82% (2,710 pounds of H2) equivalent Zr oxidation (3,300 pounds of H2 corresponding to 100% oxidation of Zr). Based on these assessments, hydrogen deflagration was assessed to result in a containment failure probability range of 2.0E-06 to 1.38E-04. The MAAP analyses presented in Section 19.2.4.4.12 indicates a range of 2,300 to 4,300 pounds, with a median of 3,300 pounds of hydrogen. This indicates a potential for 100% Zr oxidation. Given these hydrogen productions, please explain (with discussion of uncertainties) the process used to arrive at the containment failure probabilities.

Response to Question 19-146:

The explanation of the process for determining containment failure probabilities due to hydrogen production was provided in AREVA NP's response to NRC RAI No. 6, Question 19-95 dated July 7, 2008 (e-mail from Ronda Pederson (AREVA NP Inc) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 6, FSAR Ch. 19").

FSAR Impact:

The U.S. FSAR will not be changed as a result of this question.

Question 19-147:

In the Phenomenological Evaluation No. 2 – on Fuel Coolant Interactions, it is stated that EPR-specific structural response analyses to estimate the dynamic structure response of the reactor cavity/pit were not available. Instead, reference is made to work by Sehgal, et al. for structural capacity where a range of 5 to 10 kPa-s, were considered to applicable. Please:

(a) Justify the basis for the selected structural response capacity of the pit.

(b) Indicate if the current failure probability can be justified in light of "a."

Response to Question 19-147:

A response to this question will be provided by September 8, 2008.

Question 19-148:

It is possible that, in the TMI-2 accident, the instrument tubes failed from oxidation of the Zircaloy cladding, causing steam, hydrogen, and fission products to be released to the containment building before the B-loop pump was restarted (see R. E. Henry's presentation to the MAAP Users' Group on May 7, 2008). What would be the consequences of such a development in the U.S. EPR for the relevant severe accident scenarios (LOOP SS, LOOP PL, LOOP TR, and LBOP TR)? In order for us to better understand, please provide the location of hydrogen, steam, and fission product entry into the containment building, the hydrogen, steam, oxygen, and nitrogen mole fractions in the containment vs. time, the CsI, CsOH, and SrO fractions in the containment, and the containment pressure increase vs. time.

Response to Question 19-148:

A response to this question will be provided by October 3, 2008.

Question 19-149:

Please provide the following design information related to the in-core instrument tubes:

(a) A schematic of a typical in-core instrumentation tube and associated guide tube geometry.

(b) The outside guide tube diameter, material, and wall thickness.

(c) The material, the diameter of any casings where the thermocouples and/or neutron detectors are located, the wall thickness, and typical cross sectional drawing of the in-core detector and guide pipe.

(d) The material, its high-temperature yield strength as a function temperature, and its creep-rupture properties (e.g., Larson-Miller parameters) for the in-core instrument guide and other parts of the in-core detectors.

(e) A schematic of the in-core instrumentation guide tubes configuration, their location of entry into the reactor pressure vessel, and the location of entry into the instrumentation room inside the containment. This information should also show the actual location inside the containment (to be accompanied by a drawing of the containment marking the location of the instrumentation room).

(f) Description of instrumentation room compartment (room size), locations and opening areas for (flow) communication with adjacent compartments.

(g) Location of the instrument tube/guide pressure boundary between the reactor vessel and the containment building, and a discussion of failure of this boundary for both low pressure and high pressure scenarios.

Response to Question 19-149:

A response to this question will be provided by September 8, 2008.

Question 19-150:

With respect to the heat sink data provided in Table 19-47.4 of the response to RAI 19-47.5:

(a) Please provide the bottom elevation of each heat sink.

Please explain the apparent discrepancy between the values assigned to the heat sink area, and those implied by the product of length and height. (For example, in the case of HS ID 620, the heat sink is 0.04 m each in width and length but has a total area of 27.55 m2.) Please provide the actual, physical values of height, width, and area for each heat sink, if nonphysical values were listed in this table for reasons related to the computer code that was used.

Response to Question 19-150:

A response to this question will be provided by September 8, 2008.

Question 19-151:

With respect to the lumped heat sink data provided in Table 17-47-5 of the response to RAI 19-47.5:

(a) Please provide information on the material(s) making up each of the lumped heat sinks.

(b) Please explain the meaning of the indices in the column labeled “Node Containing HS(i)” and, if possible, relate them to the physical room numbers (UJAxxxxx) or to the MAAP node indices.

Response to Question 19-151:

A response to this question will be provided by September 8, 2008.

Question 19-152:

The FSAR discussion of the U.S. EPR PRA maintenance and update program does not refer to the guidance provide in Regulatory Guide 1.200 for such programs. Please provide an extended discussion that considers the Regulatory Guide material.

Response to Question 19-152:

The PRA maintenance and upgrade program described in U.S. EPR FSAR Tier 2, Section 19.1.2.4 contains the essential elements of the PRA maintenance and upgrade guidance provided in Section 1.4 of RG 1.200, including the following summary characteristics:

- Maintain PRA inputs and collect new information.
- Ensure cumulative impact of pending plant changes are considered.
- Maintain configuration control of computer codes used in the PRA.
- Identify when PRA needs to be updated based on new information or new models/techniques/tools.
- Ensure peer review is performed on PRA upgrades.

U.S. EPR FSAR Tier 2, Section 19.1.2.4.1 will be revised to add a reference to RG 1.200

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 19.1.2.4.1 will be revised as described in the response and indicated in the enclosed markup.

Question 19-153:

In Section 19.1.4.2.2.6, on page 19.1-104, the FSAR states, "… some authors have assessed containment failure." Please provide references to support this.

Response to Question 19-153:

The reference for the above statement is provided in NUREG-1524, "A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel Coolant Interaction Issues: Second Steam Explosion Review Work Group Workshop." Specifically, the summary of findings of the Second Steam Explosion Review Group (SERG2) is contained in Table E.1, Alpha-Mode Failure Probability Estimates Given a Core Melt Accident (page x). In this summary, participants Seghal of Sweden and Theofanous assessed the containment failure probability as "physically unreasonable." Additionally, participants Fauske and Henry assessed it as "vanishingly small."

A reference to NUREG-1524 will be added to U.S. EPR FSAR Tier 2, Section 19.1.4.2.2.6.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 19.1.4.2.2.6 will be revised as described in the response and indicated in the enclosed markup.

Question 19-154:

The environmental conditions presented in Section 19.2.4.4.5.1 do not appear to match the values given in the referenced Figure 19.2-20. Please explain the relation between the values cited in the text and those given in Figure 19.2-20.

Response to Question 19-154:

A response to this question will be provided by September 8, 2008.

Question 19-155:

The FSAR Section 19.1.5.3.3.7 (Fire Events L2 PRA Insights) states that the contribution to LRF related to "this phenomenon" (ISGTR) is discussed in Section 19.1.5.2.2.3. This latter subsection is a discussion of internal flooding significant L1 cutsets and sequences. Please provide a discussion of the topic consistent with other Level 2 insights for internal events at power.

Response to Question 19-155:

The reference to U.S. EPR FSAR Tier 2 Section 19.1.5.2.2.3 is incorrect; the correct reference is U.S. FSAR Tier 2 Section 19.1.5.3.2.3, Internal Fires Significant L1 cutsets and sequences. The term “this phenomenon” in U.S. FSAR Tier 2 Section 19.1.5.3.3.7 is intended to refer to fire induced seal LOCAs. The relevance of this level 1 phenomenon to the level 2 insights is explained below.

The significant contribution of the creep-induced steam generator tube rupture to the fire large release frequency (LRF) is partly due to the high contribution of seal LOCAs to the fire core damage frequency (CDF). U.S. FSAR Tier 2 Section 19.1.5.2.2.3 demonstrates why seal LOCAs are an important contributor to the fire CDF. Accordingly, core damage following a seal LOCA (0.6” or 2” equivalent LOCA) is a dominant precursor of high-temperature creep-induced SGTR.

A similar discussion regarding the contribution of ISGTR to flood LRF is provided in U.S. FSAR Tier 2 Section 19.1.5.2.3.7.

U.S. FSAR Tier 2 Section 19.1.5.3.3.7 will be revised to change the reference from Section 19.1.5.2.2.3 to Section 19.1.5.3.2.3 and to clarify that the reference is about seal LOCAs contribution to the fire CDF.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 19.1.5.3.3.7 will be revised as described in the response and indicated in the enclosed markup.

Question 19-156:

The FSAR compares the CCFP from all fire events (at power) of 0.02 to a goal of less than approximately 0.1 CCFP. A similar comparison is made for the flooding at power CCFP of 0.018. The internal event CCFP of 0.075 is also individually compared to this goal.

Regulatory Guide 1.206 notes the design should be compared against the Commission's approved use of a containment performance goal which includes a probabilistic goal that the CCFP be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA.

Please provide a discussion of the rationale for the comparison of the CCFP components noted above rather than a comparison of the composite result of all core damage sequences with the containment performance goal.

Response to Question 19-156:

The comparison of the composite result for all core damage sequences with the containment performance goal is provided in the last paragraph in U.S. EPR FSAR Tier 2, Section 19.1.8.1. The composite conditional containment failure probability (CCFP) is 0.05.

FSAR Impact:

The U.S. FSAR will not be changed as a result of this question.

Question 19-157:

While containment isolation failure is mentioned at several places in the Level 2 PRA section of the FSAR, there is no discussion of this potential path for large releases. Please provide a discussion of the probabilities associated with various sizes of unisolated paths, the source terms or categories for these possibilities.

Response to Question 19-157:

Containment Isolation failures are an important part of the Level 2 containment analysis since they are one of the mechanisms by which fission products can be released to the environment.

Containment Isolation failures are assumed be an early containment failure mode, and so it is important to characterize the size of the containment failure in order to determine whether a containment failure contributes to the large early release frequency (LERF) for the facility.

An examination of the containment release paths and flow rates through these paths was performed. This examination resulted in the determination that leakage from a single 2 inch diameter break would fall below the criteria for “Large” release, but release from two or more 2 inch lines, as well as any single line greater than 2 inch in diameter should be considered as a “Large” release.

The containment isolation (CI) fault tree was constructed and evaluated for the two top functional events for CI:

- CI Large – This top functional event contains all of the failures that could lead to a large release of fission products. In accordance with the criteria described above, leakage from any line greater than three inches in diameter or from two or more lines less than three inches in diameter is considered a “large” release.

- CI Small – contains all of the failures that could lead to a release of fission products from the containment from a single line 2 inch in diameter or less.

The large containment isolation failures are assigned to release categories (RC) RC201, 202, 203, 204, or 205, and the small containment isolation failures are assigned to RC206 as described in U.S. EPR FSAR Tier 2, Table 19.1-19.

The annual frequency of a release due to a large containment isolation failure is 9.3E-10/yr (0.2% of LRF). The annual frequency of a release due to a small containment isolation failure is 1.6E-08/yr (3.1% of LRF).

The source terms associated with these release categories are shown in the U.S. EPR FSAR Tier 2, Table 19.1-20.

FSAR Impact:

The U.S. FSAR will not be changed as a result of this question.

Question 19-158:

The first paragraph of Section 19.1.6.3.1 states:

"The analysis of shutdown conditions takes the results of the at-power Level 2 PRA and applies then, with appropriate assumptions, to the results of the shutdown PRA analysis. This approach is judged to be bounding for the low power/shutdown conditions, for both the release category frequencies and for the severity of the source terms expected from accidents initiated from low power or shutdown."

The release category frequencies are a function of frequencies of various initiating event, the system unavailabilities, the human error probabilities, and the conditional probability associated with containment failure, given core damage. All of these elements of release category frequencies are vastly different under low power and shutdown states versus at-power conditions. Furthermore, the release mechanisms and magnitudes could be vastly different for at-power as compared to low power and shutdown states. For instance, if the reactor vessel head were removed, all radionuclide releases from the fuel would be expected to enter the containment without minimal benefit from retention on reactor coolant system structures that are otherwise present in the transport path under at-power conditions. Also, under the condition for air intrusion into the fuel assemblies, enhanced oxidation can result in some of the fission products (most notably Ru) that are otherwise highly refractory, to transform into more volatile valance states. Compounded with the possibility of an open containment, it is conceivable that the most consequential release categories for at-power conditions, may not necessarily bound the radionuclide releases associated with some of the low-power and shutdown states.

(a) Please justify that the approach stated is bounding for both the release category frequencies and for the severity of the source terms.

(b) Please provide the results of EPR-specific MAAP calculations that demonstrate the applicability of at-power release categories to the shutdown states. Please include the results for hydrogen and oxygen concentrations under some of the most frequency-dominant shutdown states, for which the containment is expected to be open.

(c) Please discuss the risk implication of significantly higher Ru releases under more oxidizing conditions.

Response to Question 19-158:

A response to this question will be provided by November 4, 2008.

Question 19-159:

It is important to establish the effectiveness of the severe accident mitigation features in the U.S. EPR. Accordingly, please provide the results of the following MAAP 4.0.7 analyses of variations in the LOOP SS and LOOP TR relevant severe accident scenarios:

(a) The core melt stabilization system (CMSS) does not function but other severe accident mitigation systems work as designed.

(b) The severe accident heat removal system (SAHRS) does not function but other severe accident mitigation systems work as designed.

(c) The passive autocatalytic recombiners (PARs) do not function but other severe accident mitigation systems work as designed.

Please provide comparisons against the scenarios where all of the severe accident mitigation systems work as designed. The results should include event summaries, and plots versus time of containment pressure, mole fractions (hydrogen, oxygen, steam, nitrogen, and carbon monoxide), and concrete ablation depths (axial and radial) in the cavity and melt spreading room, for at least 24 hours after core damage.

Response to Question 19-159:

A response to this question will be provided by October 3, 2008.

Question 19-160:

In the response to RAI question 19-57, you state that it is assumed that core damage results from an uncontrolled reactivity event and an early containment failure due to overpressure from the steam produced in the core. Since the main steam line break inside containment initiator results in by far the highest contributor to the large release frequency for the U.S. EPR, it is necessary to assess its impact on containment performance over the first 24 hours after accident initiation. It is also necessary to evaluate when and how core damage might actually result, in order to facilitate the development of emergency planning and accident management procedures. Please perform a deterministic analysis of the most likely main steam line break inside containment (MSLBI) scenario with failure of I&C signals for MSIV and MFW isolation of at least three steam generators and leading to an uncontrolled reactivity event during overcooling. Please provide the results of the analysis, including an event summary and plots versus time of reactor power, primary system pressure, containment pressure, and fission product releases to the environment for at least the first 24 hours after accident initiation.

Response to Question 19-160:

A response to this question will be provided by November 4, 2008.

Question 19-161:

In order to permit comparison with the results of independent confirmatory severe accident analyses with those obtained from MAAP 4.0.7 in support of the U.S. EPR PRA, please provide results for the following MAAP calculations, if available:

a) [st_1_5bar] SBO with 0.6-in. RCP seal LOCA with successful primary-side depressurization and passive SAHRS flooding.

b) [st1.5] SBO with 0.6-in. RCP seal LOCA with successful primary-side depressurization and passive SAHRS flooding, and a 1-in. failure of containment isolation.

c) [st1_10a] SBO leading to induced hot leg rupture after core damage, and failure of SAHRS passive flooding resulting in MCCI.

d) [st2.3] SBO with 2-in. cold leg LOCA leading to induced SGTR as a result of opening all secondary-side relief valves at the time of core uncovery.

e) [st3_2a] 3-in. hot leg LOCA to the fuel building with 3/4 trains each of MHSI and LHSI available, successful primary-side depressurization, and successful passive SAHRS flooding.

Please include in the response the time-dependent graphs for the following variables, for the full duration of the calculation:

1. Pressure in the RPV, RCS, or pressurizer.

2. Temperature in the upper plenum of the RPV.

3. Actual or swollen (please specify) water level in the RPV core region and downcomer.

4. Instantaneous or cumulative flow rate through the relief and depressurization valves of the pressurizer.

5. Maximum temperature of the fuel or cladding.

6. Mass of corium debris relocated to the lower head of the reactor vessel.

7. Maximum temperature in the hot leg pipe walls (please identify the hot leg location where this maximum occurs)

8. Maximum temperature in the pressurizer wall or surge line pipe wall.

9. Maximum temperature of the steam generator tube walls.

10. Cumulative mass of H_2 generated in-vessel.

11. Cumulative mass of H_2, CO, CO_2 generated ex-vessel.

12. Pressure in each steam generator.

13. Actual or swollen (please specify) water level in the boiler and downcomer regions of each steam generator.

14. Instantaneous or cumulative flow rate through the relief and safety valves of each steam generator.

15. Pressure in the containment.

16. Atmosphere temperature in the containment.

17. Water level in the containment.

18. Axial and sidewall ablation depths due to MCCI in the reactor pit.

19. Temperature of the reactor pit melt plug.

20. Axial and sidewall ablation depths due to MCCI in the spreading area (if applicable).

21. Temperature of (ex-vessel) corium debris in the reactor pit and the spreading compartment.

22. Instantaneous or cumulative safety injection flow rate to the RCS.

23. Instantaneous or cumulative feedwater flow (normal or otherwise, please specify) rate to the steam generators.

24. Instantaneous or cumulative amount of hydrogen recombination by the PARs.

25. Mole fractions of steam, oxygen, hydrogen, CO, CO2, and N2.

26. Fraction of initial core inventory of each MAAP radionuclide class into the environment.

In addition, please provide the MAAP-predicted time of occurrence (where applicable for individual scenarios) of the following events:

27. Reactor and pump trips.

28. Gap release from the fuel.

29. First relocation of corium to the core plate.

30. First relocation of corium to the lower head of the reactor vessel.

31. Induced hot leg or steam generator tube rupture.

32. Generation of SI signal, primary- or secondary-side depressurization.

33. Reactor vessel melt-through.

34. Failure of the reactor pit melt plug and beginning of relocation to the spreading compartment.

35. Start of SAHRS passive flooding of the spreading compartment.

36. Containment failure.

Response to Question 19-161:

A response to this question will be provided by October 3, 2008.

U.S. EPR Final Safety Analysis Report Markups

The PRA model credits the alternate feed from Division 1 to Division 2 and from Division 4 to Division 3 in SBO conditions and in non-SBO conditions. This change does not modify the availability of those functions or the context in which they will be performed, but modifies the way that they will be executed.

The failure probability of those functions, as modeled in the PRA, is dominated by human errors. Those human errors were assigned HEP values judged to be conservative for this alternate feed configuration. Therefore, this design change is judged not to have a significant impact on the current conclusions of the PRA.

19.1.2.4.1 Description of PRA Maintenance and Update Program

The U.S. EPR PRA model and supporting documentation are maintained so that they continue to reflect the as-designed characteristics of the plant. Consistent with the ASME PRA Standard, Reference 5, and RG 1.200, a process is in place to perform the following as applicable to the certified design: 19-152

- Monitor PRA inputs and collect any new information relevant to the PRA.
- Maintain and upgrade the PRA to be consistent with the design.
- Consider cumulative impacts of pending changes when applying the PRA.
- Consider impacts of changes for previously implemented risk-informed decisions that used the PRA (e.g., RAP).
- Maintain configuration control of the computational methods used to support the PRA.
- Document the PRA model and processes.

When reviewing pending design changes and proposed model improvements, the impact on the CDF and LRF are estimated. Based on the estimated impact, one of the following update approaches will be taken:

- If the effect of the change(s) is judged to be safety significant, a PRA model update is implemented commensurate with the safety significance of the pending change without waiting for the routine update cycle. The significance of a pending change(s) and the urgency of its implementation within the PRA model are based on consideration of several change attributes including the level of complexity of the change, the ability to manage/control potential cumulative modeling impacts, and the estimated quantitative impact to the model risk results. The quantitative results could increase or decrease depending on the specific design change. However, in general, if a change(s) is estimated to increase the CDF to greater than 1.0E-06/yr, the model would be updated in a timely manner without waiting for the routine update cycle.
- If the effect of the change is judged not significant, the PRA model will be revised at the next scheduled update.

Since the LRF results are dominated by the SLBI sequence discussed in Section 19.1.4.2.2.2, Section 19.1.4.2.2.4, and Section 19.1.4.2.2.5 and SGTR sequences (initiated in Level 1), no individual phenomenological events make a large enough contribution to LRF for these to lead to a significant reduction in LRF when set equal to zero.

The following events can lead to a significant increase in LRF if set equal to 1:

- Hydrogen combustion related basic events for failure of the containment due to deflagration prior to vessel failure (L2PH VECF-H2DEF(HL) – deflagration fails containment after hot leg rupture. If assumed to always occur this event would lead to a seven times increase in LRF.
- Hydrogen combustion related basic events for failure of the containment due to loads from accelerated flames prior to vessel failure (L2PH VECF-FA(H) and L2PH VECF-FA(HL) – discussed in Section 19.1.4.2.2.4). If assumed to always occur, these events would lead to an 11 times or 9 times increase in LRF, respectively.
- The event L2PH STM EXP INV LP (containment failure due to in-vessel steam explosion), would, if assumed to always occur, lead to nearly a three-fold increase in LRF.

It can be noted that deflagration causing failure of the containment is close to being a physically unreasonable event. Its base probability of 1.38E-04 in case of hot leg rupture was assessed with some degree of conservatism. The analysis was based on upper bound (top of range of uncertainty) values for the masses of hydrogen present in containment rather than performing detailed Monte Carlo simulation as was performed for some other events, and no credit was taken for consumption of hydrogen due to benign burning.

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Similarly, it is also noted that some authors have assessed containment failure due to steam explosion as a physically unreasonable event~~.~~—refer to NUREG-1524 (Reference 47). The U.S. EPR Level 2 analysis also assessed this as a very low probability event, but with an assessed probability greater than 1E-06, it was not judged to be of sufficiently low probability for it to be removed from the model. Sensitivity to this event arises because, if it is not excluded from the model, it is applicable to a large proportion of core damage sequences.

Thermally-induced steam generator sequences do not play a significant role in LRF for internal events. However, given that sequences with a depressurized secondary side contribute nine percent of CDF, sensitivity studies were undertaken to study the factors influencing this contribution. The sensitivity to manual depressurization and availability of feedwater was therefore studied. It was found that, for the case of internal events, unavailability of primary depressurization had a larger impact on the frequency of RC702 than unavailability of feedwater. However, while the combined

19.1.5.3.3.7 Fire Events Level 2 PRA Insights

In the absence of the specific challenges and bypasses of containment seen in the internal events analysis, the results for LRF for fire events are dominated by severe accident phenomenological issues. The specific issue for fires is the possibility of an accelerated flame arising from hydrogen combustion in the lower or middle equipment rooms during the in-vessel phase of a high pressure core melt. Further background discussion on the analysis of this issue is provided in Section 19.1.4.2.2.4.

As also discussed in Section 19.1.4.2.2.4 for internal events, sequences involving containment failure due to loads from an accelerated flame originating in the lower, middle or upper equipment rooms prior to vessel failure are visible contributors to LRF. The key features and assumptions of the analysis of accelerated flames are discussed in Section 19.1.4.2.2.4 and not repeated here.

19-155

The phenomena of thermally-induced steam generator tube rupture, which was assessed as having a large probability for equivalent two-inch seal LOCAs ~~(seal or otherwise)~~ in conjunction with a depressurized secondary side and an absence of feedwater to the SGs, also features in the results (i.e., 13 percent contribution to LRF). Seal LOCAs are a contributor to the fire CDF, ~~The contribution of this phenomenon is~~as discussed in ~~Section 19.1.5.2.2.3~~Section 19.1.5.3.2.3. Sensitivity studies show~~ed~~ that LRF did not significantly increase due to this phenomenon even in the bounding case of assumed concurrent unavailability of feedwater and depressurization functions.

Despite the dominance of a single phenomenological issue for LRF, it is noted that LRF is only approximately two percent of the CDF for fire events.

Other phenomenological challenges were not identified as leading to significant probabilities of large release.

19.1.5.4 Other Externals Risk Evaluation

The design certification scope of external event screening includes an assessment of high winds and tornadoes and external flooding as described below.

A COL applicant that references the U.S. EPR design certification will perform the site-specific external event screening analysis for external events applicable to their site.

19.1.5.4.1 High Winds and Tornado Risk Evaluation

All U.S. EPR Seismic Category I structures are designed to meet the following standards for high winds and tornadoes.

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47. NUREG-1524, "A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel Coolant Interaction Issues: Second Steam Explosion Review Work Group Workshop," U.S. Nuclear Regulatory Commission, 1996.