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

QUESTIONS

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 percent (1,580 pounds of H2) and 82 percent (2,710 pounds of H2) equivalent Zr oxidation (3,300 pounds of H2 corresponding to 100 percent 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 percent Zr oxidation. Given these hydrogen productions, please explain (with discussion of uncertainties) the process used to arrive at the containment failure probabilities.

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”.

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.

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.

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.

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.

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.

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.

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.

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.

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.

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.

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.

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.

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 is 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.

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 one-inch 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 two-inch 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] three-inch 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, CO_2, and N_2.

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.