Request for Additional Information No. 133 (1456), Revision 0

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U. S. EPR Standard Design Certification
AREVA NP Inc.
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SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 2 (ESBWR/ABWR Projects) (SPLB)

19-229

(Intentionally deleted.)

19-230

(Follow-up to Question 19-147) In response to Question 19-147, the applicant provided a subjective probability distribution for structural capacity of the EPR reactor pit for dynamic pressure loads. Table 19-147-1 shows that the reactor pit is expected to fail with certainty for loads exceeding 20 kPa-s. NRC assessment of ex-vessel steam explosion loads (NUREG/CR-6849) under similar conditions show maximum loads resulting from FCI energetics ranging from about as low as 10 to as high as a several hundred kPa-s, depending on the melt pour and analysis assumptions and conditions.

(a) The approach described in the response appears to be subjective. One acceptable approach to this problem is to determine the threshold impulse load at which the pit structure will have zero probability of failure (i.e., this approach is typically considered as bounding. For instance, in other recent submittals, the DYNA 3D model was used to establish the impulse threshold for the portion of containment that was subject to steam explosion-induced impulse loads). Please perform a mechanistic analysis that supports the assigned uncertainty distribution.

(b) Please discuss the implication of the NUREG/CR-6849 results for U.S. EPR in light of the assumed reactor pit capacity.

(c) Please provide the technical justification for arriving at ex-vessel FCI loads that are much lower than has been estimated for other plants under similar conditions (e.g. AP1000). This should include plant-specific analysis using methods that are similar to those that are being used in other contemporary studies (e.g. see Westinghouse AP1000 DCD, GEH ESBWR DCD).

(d) Please provide the range of expected loads on the RPV, and if there is any potential for RPV uplift impacting containment penetrations.

(e) Please provide an analysis of the impact of the reactor pit failure on severe accident progression for U.S. EPR.

19-231

(Follow-up to Question 19-189) In response to Question 19-189, the applicant discusses material testing that has been performed for the U.S. EPR Zirconia bricks, including erosion tests for contact with oxidic molten core debris.

Please provide the results of these tests, including the specification of the stabilized Zirconia that will be used for U.S. EPR versus those tested.

19-232

(Follow-up to Question 19-84d) The response to Question 19-84d discusses the results of sensitivity cases that were investigated for source term calculations.

a. In discussions of the "effect of the SAHRS" (severe accident heat removal system), "isolation failure size," and "small leakage," the text refers to Figures 19-84d-1 through 19-84d-6 that are missing in the response. Please, provide these figures.

b. In discussions under "small leakage," the containment leak rate for 2- and 3-inch failure diameter (Figure 19-84-9d [SIC]), the response indicates that the flow rate for the 3-inch diameter case is conservatively calculated as (9/4) x (the leak rate for a 2-inch diameter). However, the figure illustrating the results show an inconsistency, namely, where the leak rate is lower for the 3-inch diameter case as compared to that for the 2-inch diameter break. Please, explain this apparent inconsistency.

c. In addition, in discussing the effects of molten corium-to-concrete interaction (MCCI), the response refers to Figures 19-84d-11 and 13 depicting lower values of SrO releases for cases involving MCCI as compared to accidents where MCCI is prevented. Please explain the phenomenological processes causing this apparent anomaly.

19-233

(Follow-up to Question 19-84e) The response to Question 19-84e discusses the assumptions applicable to analysis of steam generator tube rupture (SGTR) source terms (i.e., Release Category 702). The response confirms that the analysis assumes the failure of a single tube, whereas the results of a preliminary sensitivity study by AREVA shows the increase in fission product source term under the postulated condition of more than a single tube rupture. Please address the likelihood for multiple SGTRs and provide associated source terms.

19-234

(Follow-up to Question 19-84c) The response to Question 19-84c indicates alarm timing errors that were used in the consequence analysis, and provides a table comparing the results for “Base” and “All.”   The error in alarm timing affects many release categories. Please provide an explanation of the meaning of “Base” and “All”.

19-235

(Follow-up to Question 19-114) The response to Question 19-114 discusses flame acceleration and analysis approach on the uncertainty of zirconium oxidation. The method relied on realistic values of hydrogen production, which were lower than 100% zirconium oxidation. Please explain the potential for flame acceleration and detonation, if one were to consider hydrogen masses corresponding to 100% of the equivalent in-vessel zirconium oxidation (see 10 CFR 50.44(c).

19-236

(Follow-up to Question 19-121b) The response to Question 19-121b discusses the effect of using the mean core damage frequency (CDF) as opposed to using point estimate CDF value for cost-benefit analysis of SAMDAs, and it indicates a potential increase in maximum benefit of a factor of 8 if the mean CDF values were used. The NRC guidance, as well as the American Society of Mechanical Engineers (ASME) PRA standard, specifically requests mean values rather than point estimates (see Regulatory Guide 1.174, Section 2.2.5.5; Regulatory Guide 1.206, Section C.I.19, Appendix A; SRP 19.0, Section III; ASME RA-Sb-2005, Supporting Requirement QUA2b). Therefore, please explain why the mean CDF was not used in the evaluation.

19-237

(Follow-up to Question 19-121c) The response to Question 19-121c discusses five sensitivity analyses that were performed to assess the uncertainties in the maximum benefit calculations. The calculations were performed on a segregated basis; i.e., considering individual effects one at a time. This method would not provide the true effects of combination of uncertain inputs. For example, updating the constant, representing a string of replacement power costs provided in terms of “1993 dollars” in the NUREG/BR-0184, to represent a value in “2008 dollars” is not a sensitivity analysis, and perhaps needs to be used as the basic input for the analysis.

In addition, the present value of the replacement power for a single event of $1.4E+9 (in “1993 dollars”) with a discount rate of 3 percent in the NUREG/BR-0184 corresponds to an average reactor life of 24 years. This value should be adjusted for the reactor life of 60 years, which is applicable to the U.S. EPR. Therefore, please explain why a consolidated calculation using the upper bound estimates for the onsite clean up cost, the onsite plant personnel dose, and the inflation adjusted replacement power costs in combination with a 3 percent discount rate (as recommended in NUREG/BR-0058, and used in the ANP-10290) was not performed to estimate the maximum benefit. Note that the final results should consider the mean CDF as discussed in the follow–up to Question 19-121b.

19-238

(Follow-up to Question 19-121f) The response to Question 19-121f indicates that if the released water/steam through the safety relief valve following a SGTR accident were to be injected (through a specially designed system) back into the containment, the resulting containment pressure would be comparable with containment loads for other severe accident scenarios that have been analyzed (see Figure 19-121-1 and 19-121-3). In fact, the associated loads both in terms of pressure and water loading on the containment should be similar to that of a small steam line break, with a known release path to a specially designed in-containment quench/scrub system. Therefore, at least from the safety point of view, this appears to be a viable alternative that should have been considered as part of the SAMDA cost benefit studies. Please perform an analysis of this SAMDA to determine if it merits implementation into the design, or explain why it is not necessary to do so.

19-239

(Follow-up to Question 19-157) The response to Question 19-157 discusses containment isolation failure and its contribution to large early release. While the split between large and small isolation failures is deducible from careful reading of the response, the bulleted specifications for the top events conflict with the rest of the text and would benefit from some rewriting. The top event "CI Large" description in the first bulleted item, as written, defines leakage from two or more lines (of any size however small) as a large release. The bands for single line leaks are not contiguous (currently 2" or less for CI Small and greater than 3" for CI Large for single line leaks.)

The response indicates that the annual frequency of release due to large containment isolation failures (defined as release category [RC] 201—205) is 9.30E-10, contributing 0.2 percent of LRF. This LRF frequency is the sum of frequencies from internal events (internal, fire and floods) as given in Tables 19.1-24, 19.1-50, and 19.1-75. The total LRF value from these tables is 2.67E-08, which makes contribution from large containment isolation failures to be about 3.5 percent of the internal events LRF, and not 0.2 percent as indicated in the response. Please clarify whether the total LRF values for internal events have changed, if so please provide the new and revised estimates and please update the affected tables.

In addition, in the response to Question 19-84a, RC 206 is identified as large early release, and considered as consequence dominant for Level 3 analysis. However, Table 19.1-24 of the Final Safety Analysis Report (FSAR), neither identifies this release category as large release, nor provides its frequency. Please clarify this inconsistency, and if RC 206 is considered as part of LRF, then the affected FSAR tables for internal events need to be revised.

19-240

(Follow-up to Question 19-79)

1. Please confirm the relationship for the stress and Larson-Miller parameter given in Table 19-79-1 of the response to RAI 19-79. Specifically, should LMP be multiplied by the fitting parameter bfit or added to it?
2. More information is needed to clarify the response to question 19-79a, as summarized below.

- Since cases where the secondary side is depressurized dominate the risk in current PWRs, please run a variation of the MAAP base case with degraded tubes where the steam generators are depressurized and report the damage fractions and the times of predicted hot leg failure, tube failure, and when the core outlet temperature reaches 650 C. Also provide plots of the steam generator tube temperatures.

- If there are any relevant PRA sequences that would include the 2" diameter breaks, please identify them, and state their core damage frequencies. For MAAP case 1.1i (depressurized SGs), tube failure is estimated to occur about 25 minutes after the core outlet temperature reaches 650 C (11918 sec. vs. 10440 sec.). How does this relatively short time interval affect potential severe accident management high-level actions and strategies? What would be the tube failure times for variations of this case where tubes are damaged, in particular for 1/2, 2/3, and 3/4 through-wall volumetric degradation (MAAP parameter FERSGT=0.5, 0.33 and 0.25). Such damage can realistically be expected to occur from foreign object wear above the top of the tube sheet.

3. The response to Question 19-79e states that system-related top events in the containment event trees are used to model the status of containment isolation, safety injection, severe accident heat removal, and other systems. But FSAR Section 19.1.4.2.1.3 states that the CDES are not, however, directly transferred to Level 2 CETs. Rather, each individual end state is transferred through an intermediate event tree, referred to as CDES link event tree, prior to transfer to a Level 2 CET. Please provide the details of these link event trees, including diagrams, tables of grouping and quantifications, how they are applied to each of the CDESs and how they are linked to the Level 2 CETs that are shown in FSAR Appendix 19C.

19-241

Please provide the inner diameter of the control rod guide assembly (CRGA) in the location where it penetrates the RPV upper head.

19-242

(Follow-up to Question 19-150) With respect to the distributed heat sink data provided in Table 19-150-1 of the response to RAI 19-150:

a. Please provide the thickness of each heat sink.

It is stated in the response that the heat sink data in Table 19-150-1 are used as the source for the MAAP input deck. Pease clarify the reasons for differences in the total surface areas of various heat structure material as calculated using the data in Table 19-150-1 and those of the MAAP input deck A.

19-243

(Follow-up to Question 19-192) It is stated in the response to Question 19-192 that AREVA does not understand the regulatory basis of the question. This basis is described in NEI 91-04 (originally NUMARC 91-04), "Severe Accident Issue Closure

Guidelines," which summarizes the important information pertaining to the agreement between the nuclear industry and the NRC on severe accident management, and contains the binding implementing guidance in Section 5. As such, it is an industry initiative that was endorsed by the NRC.

The current industry technical basis stems from EPRI's Severe Accident Management Technical Basis Report (EPRI TR-101869), which was used in developing vendor-specific guidance. This guidance was provided to the NRC by the various owners groups and constitutes the technical basis of severe accident management for existing plants. What the NRC staff is now looking for is the revised technical bases for the new plants, which would consider the new features for accident prevention and mitigation. The staff needs to review this material before any COL is issued to ensure that no unexpected problems arise that would be difficult to resolve after the fact.

After a COL is issued, it is imperative that the new plants also have their SAMG implementation in place, including procedures and training, prior to initial fuel load.

The purposes of Question 19-192 were:

1. To obtain recommendations from AREVA, for the various accident scenarios being considered, regarding how best to prevent the accidents from progressing to core damage, terminate core damage once it begins, maintain the capability of the containment as long as possible, and minimize on-site and off-site releases and their effects.
2. To request AREVA to identify a COL action item that would require each COL applicant to provide documentation of the severe accident technical basis.

The NRC staff is not asking for specific severe accident management guidelines. Instead, the staff needs to evaluate the technical bases that would support the guidelines. Specifically, the staff needs to review the results obtained during the first two phases identified in the last paragraph of the initial response to question 19-192. In that regard, please provide documentation of the results of these first two phases, culminating in the overall mitigation strategies that would be developed based on symptoms and phenomena. Also provide any available supplemental analyses that demonstrate the effectiveness of the operator actions to the plant response. Please provide, to the extent practicable, a discussion of the accident sequences considered and the results of the supporting MAAP 4.0.7 analyses. Finally, please provide the necessary COL action item or a justification for not providing one.

19-244

(Follow-up to Question 19-148) The response to Question 19-148 contained results for a single instrument tube failure. The NRC Staff is also interested in what would be the consequences of failing multiple instrument tubes, from oxidation of the Zircaloy cladding.

1. Please provide an analysis of the consequences of failing <u>all</u> of the AMS probes in the region of the core where the Zircaloy oxidation takes place, for each of the relevant scenarios. Discuss how the hydrogen, steam, and fission products

would then flow through the AMS probes and out into the instrumentation rooms. Please identify the recipient containment node in the MAAP model, and discuss the potential for a large hydrogen deflagration or detonation there and in adjacent compartments. In addition, please describe the gas flows from there into the rest of the containment building. Please present similar plots (mole fractions of hydrogen, steam, oxygen, and nitrogen; containment pressures, and temperatures; and fission product releases into the containment) to those presented in the initial response to the question.

2. For both the single probe and multiple probe failure cases, provide plots of the core-to-upper plenum natural circulation, countercurrent natural circulation flow rates in the hot leg and steam generator tubes, and damage fractions in the hot leg and steam generator tubes.

19-245

The most recent revision to the severe accident source term presents the results of the MAAP cases that were re-calculated in order to address core inventory input error that was discovered by AREVA (WebCAP 2008-3905-CR and 2008-4034-CR).

Please address the following questions:

(1) Do these revised source term results require any modification to previously provided responses to RAIs by AREVA? If this is the case, please provide a list of the affected RAI responses, and resubmit your responses.

(2) Are there any other significant analyses that have been revised as a result of the revised source term data? If so, please provide a list of the affected analyses.

(3) Does this revision affect the information that has been transmitted by AREVA to the current COL applicants? Please elaborate.

19-246

Please provide detailed results of the MAAP 4.0.7 and S-RELAP5 calculations to determine Level 1 success criteria. Also include the results of benchmarking MAAP 4.0.7 against S-RELAP5 and explain how the benchmarking was performed. As part of your discussion:

1. Please provide results of all of the analytical comparisons made, and include any cases that provide a direct core heatup comparison.
2. Please explain why phenomena associated with these events are expected to extend to the remaining events of interest, such as steam generator tube rupture, steam line breaks (inside or outside containment), and loss of off site power.
3. Please list and discuss the key assumptions made either in model development activities or in processing the individual calculations.
4. Please explain why an alternative acceptance criterion of a MAAP 4.0.7 calculation of 1800 F for peak cladding temperature should be used, and why calculations

yielding peak cladding temperatures between 1400 F and 1800 F should be evaluated qualitatively or by an independent calculation before being accepted.

5. Please list and discuss the major conclusions from the comparison.
6. Please present the success criteria results obtained with MAAP 4.0.7 for all of the events considered.