Request for Additional Information No. 17, Revision 0

6/18/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 11.01 - Source Terms
SRP Section: 15.00.03 - Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors
Application Section: FSAR 11.1 and 15.0.3
RSAC Branch

QUESTIONS

11.01-1

Which dose conversion factors (DCFs) from FGR-11 Table 2.1 were used to calculate dose equivalent iodine-131 (DE I-131) nuclide-specific reactor coolant concentrations; committed effective dose equivalent (CEDE) or thyroid? Justify which DCFs were used.

11.01-2

Were effective dose equivalent or organ dose equivalent dose conversion factors from FGR-12 Table III.1 used to calculate dose equivalent Xe-133 (DE Xe-133) nuclide-specific reactor coolant concentrations? Justify which DCFs were used.

11.01-3

Are the RCS primary coolant iodine concentrations and noble gas concentrations listed in Table 11.1-2 and Table 15.0-15 adjusted to reflect 1 microCi/g DE I-131 and 210 microCi/g DE Xe-131, respectively?

15.00.03-1

The ORIGEN2.1 generation and depletion code was used to calculate the core radionuclide inventory. Oak Ridge National Laboratory (ORNL) does not support the ORIGEN2 code any longer, but instead recommends use of the ORIGEN-ARP or ORIGEN-S code included in the SCALE code package, which is kept up-to-date. SCALE 5.1 is the latest release and includes libraries for high burnup fuel, up to 72 gigawatt days per metric ton uranium (GWD/MTU). Please justify the use of an older unsupported version of the ORIGEN code.

15.00.03-2

FSAR Section 15.0.3.3.4 states that the fuel burnup ranges between approximately 5 and 62 GWD/MTU. Are the cross-section libraries used in the calculation of the core radionuclide inventory applicable to the maximum fuel burnup assumed?

15.00.03-3

Provide the length of the operating cycle and the number of cycles that are assumed in the core radionuclide inventory calculations.

15.00.03-4

Describe the use of the Murphy-Campe model in the loss-of-coolant accident (LOCA) control room dose analysis.

15.00.03-5

Provide the basis for the deep dose equivalent (DDE) reduction of 15 percent for the LOCA control room dose analysis.

15.00.03-6

In the control room dose calculations for each of the design basis accident (DBAs), you have calculated a composite control room atmospheric dispersion factor (X/Q) to account for the different locations for the unfiltered inleakage and the filtered intake. Please describe how the composite X/Qs were calculated.

15.00.03-7

In the tables listing the composite control room X/Qs for each DBA, filter bypass fractions are listed. Provide the methodology for using the filter bypass fraction in the dose calculations.

15.00.03-8

For the control room direct dose model, provide the calculations and justify the assumptions and inputs, including the determination that only the filter shine dose is non-negligible. Indicate the receptor location for the direct dose in the control room and describe what that location was chosen.

15.00.03-9

FSAR 15.0.3.4.1 states that the technical support center (TSC) is within the main control room pressure boundary and therefore has the same habitability. The control room envelope (CRE) is served by the control room emergency filtration (CREF) system. Confirm that the TSC continues to be within the CRE and served by the CREF along with the main control room (MCR) during the emergency mode of operation for each DBA.

15.00.03-10

In FSAR 15.0.3.5, the description of small line break scenario 2, break of a Chemical and Volume Control System (CVCS) line, states that no flashing occurs, and Table 15.0-21 indicates that the temperature of the coolant released is less than 212 degrees F. Regulatory Guide 1.183, Appendix A, gives guidance on assumptions on engineered safeguard feature (ESF) system leakage for the LOCA, which can be reasonably applied

to the iodine airborne release from liquid leakage from a small line break. RG 1.183, Appendix A, Section 5.5 states that if the temperature of the leakage is less than 212 degrees F or if the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the sump (or source of leakage) pH history and area ventilation rates. Please provide the airborne iodine assumption, and if it is less than 10% of the total in the leakage, please justify the assumption.

15.00.03-11

For the steam generator tube rupture (SGTR) accident, provide the duration of the releases from the unaffected and ruptured steam generators, and the basis for the assumption. FSAR Table 15.0-27 appears to give the durations as 8 hours for the unaffected SGs and 0.8 hours for the ruptured SG. Provide the bases for these release durations.

15.00.03-12

In regard to the SGTR dose analysis, please verify that the halogen and alkali metal decontamination factor assumed for the condenser is only applied prior to the manual reactor trip and loss of offsite power (LOOP) at 30 minutes into the event.

15.00.03-13

With regard to the main steam line break (MSLB) accident dose analysis, provide the basis for the assumed primary-to-secondary leak rate of 0.125 gpm/SG.

15.00.03-14

In the calculation of the amount of fuel melt that would equal approximately 90% of the dose acceptance criteria for the MSLB, provide the assumed cladding damage due to the conditions in the core that may occur in fuel that does not melt. Please justify the assumptions for the calculation of the fuel melt.

15.00.03-15

In regard to the MSLB dose analysis, assure the steaming rate of 113 lbm/sec is constant over the duration of the release. Provide the basis for this assumed steaming rate. Confirm that steaming rate includes the unaffected steam generators (SGs) and the steaming through the affected SG (SG 4).

15.00.03-16

In regard to the rod ejection accident (REA) dose analysis, provide the basis for the assumed fuel overheat (melting) fraction of 0.4 percent of fuel that experiences clad failure.

15.00.03-17

In regard to the REA dose analysis, discuss and justify whether the methodology for calculating the clad failure and fuel overheat accounts for the transient fission product release, as discussed in SRP 4.2, Revision 3, Appendix B.

15.00.03-18

The text in FSAR 15.0.3.9 for the REA dose analysis does not discuss the aerosol natural deposition in containment (as noted in FSAR Tables 15.0-40 and 15.0-41) in detail. Identify the REA analysis cases that assumed aerosol deposition.

15.00.03-19

The text in FSAR 15.0.3.9 for the REA dose analysis does not discuss elemental iodine removal. FSAR Table 15.0-40 states that the elemental iodine removal coefficients are the same as for aerosols, except limited to a decontamination factor (DF) of 100. Please justify use of models and assumptions on iodine elemental iodine removal. Provide the mechanism on which the elemental iodine removal is based. Document that this mechanism has been previously approved in reactor licensing. Identify the cases for which elemental iodine removal was assumed. Provide the time after the onset of the REA that the DF equals 100.

15.00.03-20

In regard to the fuel handling accident (FHA) dose analysis, provide the basis for assuming all the rods in the dropped assembly have failed. Assure that the mechanical failure of the EPR fuel been analyzed.

15.00.03-21

In regard to the FHA dose analysis, provide the methodology for assuming that the release location at the base of the main stack gives bounding control room atmospheric dispersion factors and therefore doses for any open airlock, equipment hatch, containment penetration or fuel building opening.

15.00.03-22

Provide the definition of "recently irradiated fuel" as used in the EPR technical specifications, including the duration that the fuel has undergone decay. Provide the location for this definiton.

15.00.03-23

Provide assurance that you have performed a dose analysis of the FHA for fuel that has less than 34 hours of decay time, using the containment or fuel building closure requirements in the Technical Specifications. If not, provide the means of preventing fuel movement prior to 34 hours decay time.

15.00.03-24

In the LOCA and REA dose analyses, the Powers natural deposition model and the "Henry model" for aerosol removal in containment were combined to model aerosol deposition in the containment. The Henry model has not previously been reviewed or found acceptable for use in reactor licensing. Please justify the use of the Henry model for aerosol natural deposition in the EPR containment.

15.00.03-25

In the LOCA and REA dose analyses, aerosol deposition in containment was modeled in part by using the 10th percentile Powers natural deposition correlation in RADTRAD. Considering that the Powers natural deposition correlation was developed using operating pressurized water reactor (PWR) and boiling water reactor (BWR) information on containment geometry and power, justify the applicability of the Powers natural deposition correlation to the EPR containment.

15.00.03-26

In regard to the LOCA and REA dose analyses, provide the methodology for combining the Powers model and the Henry model to give the effective aerosol removal coefficients in FSAR Tables 15.0-52 and 15.0-41. Include the justification for using these models in combination.

15.00.03-27

In regard to the LOCA dose analysis, FSAR Table 15.0-52 states that the elemental iodine removal coefficients are the same as for aerosols, except limited to a decontamination factor (DF) of 100. Please justify use of models and assumptions on iodine elemental iodine removal. Provide the methodology used for elemental iodine removal. Assure that this methodology been previously approved for reactor licensing. Identify the cases where elemental iodine removal is assumed. Provide the time after the onset of the LOCA that the DF equals 100.

15.00.03-28

For the LOCA dose analysis containment leakage pathway, provide the amount of mixing assumed in the annulus volume after drawdown of the shield building. Include a discussion and justification for the mixing assumed.

15.00.03-29

In regard to the LOCA containment leakage pathway dose analysis, explain the change in release location during annulus drawdown as compared to the location during purge and after drawdown.

15.00.03-30

In regard to the LOCA engineered safeguard features (ESF) component leakage dose analysis, provide the ESF component leakage source term and the basis for the assumed values.

15.00.03-31

For the LOCA ESF component leakage pathway, provide the amount of mixing assumed in the safeguard building and fuel building after drawdown. Please provide a discussion and justification for the mixing.

15.00.03-32

Explain the change in release location during safeguard building and fuel building drawdown vs. after drawdown for the LOCA ESF component leakage pathway dose analysis.