Response to

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: 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

Question 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.

Response to Question 11.01-1:

The CEDE values in Federal Guidance Report No. 11 (FGR-11) Table 2.1 are used as the DCFs to calculate DE I-131 nuclide-specific reactor coolant concentrations. Selection of these DCFs is based on the dose acceptance criterion for the alternative source term, namely the total effective dose equivalent, which consists of two components: the CEDE for the inhalation pathway and the effective dose equivalent, also referred to as the deep dose equivalent, for external radiation exposure.

FSAR Impact:

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

Question 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.

Response to Question 11.01-2:

The effective dose equivalent values in Federal Guidance Report No. 12 (FGR-12) Table III.1 are used as the dose conversion factors (DCF) to calculate DE Xe-133 nuclide-specific reactor coolant concentrations. Selection of these DCFs is based on the dose acceptance criterion for the alternative source term, namely the total effective dose equivalent, which consists of two components: the committed effective dose equivalent for the inhalation pathway and the effective dose equivalent for external radiation exposure.

FSAR Impact:

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

Question 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?

Response to Question 11.01-3:

The reactor coolant system primary coolant concentrations of individual radionuclides presented in U.S. EPR FSAR, Tier 2 Table 11.1-2 and Table 15.0-15 include the corresponding DE Xe-133 and DE I-131 concentrations, specifically, 210 microCi/g DE Xe-133 and 1 microCi/g DE I-131. These radionuclide concentrations are used in the analysis of the Chapter 15 design basis accidents involving primary coolant releases.

Although otherwise identical, the values presented in Table 11.1-2 were rounded to two significant figures, while the same values presented in Table 15.0-15 were rounded to three significant figures.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-1:

Regulatory Guide 1.183 (RG 1.183), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," recommends the use of the ORIGEN-2 computer code for the generation and depletion of the core inventory. Therefore, AREVA NP used ORIGEN-2.1 in the calculation of the design basis core radionuclide inventory.

A core radionuclide inventory comparison between ORIGEN-2.1 and ORIGEN-S was performed to evaluate the effect on loss of coolant accident (LOCA) dose for the containment leakage pathway. The sensitivity of the two source terms was also evaluated for different groupings of principal radionuclides (84 and 294 in number) used in the analysis of design basis accidents. In the comparison, the core enrichment and burnup are assumed to be 5 percent U-235 and 62 GWD/MTU, respectively, and the low population zone (LPZ) dose was calculated.

The results of the evaluation are presented in Table 15.00.03-1-1 and demonstrate that the application of ORIGEN-2.1 and the use of the 84 radionuclides selected are appropriate based upon the relative agreement between the radiological doses calculated using the different versions of the ORIGEN code. As shown in Table 15.00.03-1-1, the ORIGEN-2 dose calculations are more conservative than the ORIGEN-S dose calculations.

Table 15.00.03-1-1—ORIGEN Code Comparison

Number of Radionuclides	LOCA Containment Leakage Pathway – LPZ TEDE Dose (rem)		Dose Ratio ORIGEN-2/ORIGEN-S
	ORIGEN-2	ORIGEN-S	
84	6.165	5.472	1.13
294	6.170	5.476	1.13
Dose differential (84 vs. 294 nuclides)	0.005	0.004	

FSAR Impact:

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

Question 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?

Response to Question 15.00.03-2:

In RG 1.183, Section 3.1, 62 gigawatt days per metric ton uranium (GWD/MTU) is identified as the peak burnup acceptable for the gap fractions in Table 3 of the RG. The response to RAI Question 15.00.03-1 demonstrates that the ORIGEN-2.1 code with the PWR-UE cross section libraries is acceptable for the maximum burnup assumed (i.e., 62 GWD/MTU).

ORIGEN-2, Version 2.1, with the PWR-UE cross-section data libraries for extended burnup (from ORNL/TM-11018) are used to calculate the core radionuclide inventory. These cross-section libraries are based on an updated reactor model for PWRs operating on uranium fuel cycles, using recent (i.e., 1989) data on fuel cycle schemes and reactor core designs. The cross-section libraries were extrapolated to 62 GWD/MTU, the maximum fuel burnup assumed.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-3:

The core radionuclide inventory calculations conservatively assume continuous, full-power operation from initial startup (i.e., no exposure) to 62 gigawatt days per metric ton uranium (GWD/MTU) burnup with no outages or power reductions. Therefore, in the core radionuclide inventory calculations, the number and length of operating cycles is immaterial.

FSAR Impact:

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

Question 15.00.03-4:

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

Response to Question 15.00.03-4:

The Murphy/Campe model used in the main control room LOCA deep dose equivalent (DDE) calculations is in accordance with the guidance of RG 1.183, Section 4.2.7. The following expression (from RG 1.183, Section 4.2.7) is used to correct the semi-infinite DDE to a finite cloud dose, DDE_{finite}:

$$DDE_{finite} = DDE * V^{0.338} / 1173 \qquad \text{(Eq. 1)}$$

where V is the control room volume (ft^3) modeled as a hemisphere.

FSAR Impact:

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

Question 15.00.03-5:

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

Response to Question 15.00.03-5:

The 15 percent reduction in the loss of coolant accident (LOCA) DDE dose to main control room (MCR) personnel is calculated using the following expression:

$$DRF = (DDE_{finite1} / DDE_{finite2}) = (V_1/V_2)^{0.338} = (200{,}000 / 133{,}000)^{0.338} = 1.15 \qquad \text{(Eq. 1)}$$

where DRF is the dose reduction factor, V_1 is the mixing volume of the entire MCR pressure boundary (200,000 ft^3 on elevations +53 ft and +69 ft), and V_2 is the volume of the MCR proper (133,000 ft^3, on elevation +53 ft). This expression is based on the Murphy/Campe model described in the response to RAI Question 15.00.03-4 and also in RG 1.183.

The MCR volume on Elevation +69 ft (67,000 ft^3) houses the Safeguard Buildings' air supply, filtration and air conditioning systems. The DDE dose to MCR occupants from radiation emanating from this volume is minimal because of the ≈20 inch thick concrete floor separating the two MCR volumes.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-6:

The analytical model employed in the main control room (MCR) radiological evaluations allows for only one intake flow to the MCR, with partial or full filtration. To account for the two pathways (i.e., a primary intake flow from one location, with and without filtration, and a secondary unfiltered intake flow from a different location), it is necessary to define effective atmospheric dispersion factors and associated MCR intake-filter bypass fractions. These are defined as follows, for each time interval of interest:

$$(\chi/Q)_{eff} = [F_{intake}\ (\chi/Q)_{intake} + F_{leakage}\ (\chi/Q)_{leakage}] / [F_{intake} + F_{leakage}] \quad \text{(Eq. 1)}$$

$$f_{bypass} = F_{leakage}\ (\chi/Q)_{leakage} / [F_{intake}\ (\chi/Q)_{intake} + F_{leakage}\ (\chi/Q)_{leakage}] \quad \text{(Eq. 2)}$$

where:

$(\chi/Q)_{eff}$ = effective atmospheric dispersion factor (sec/m^3), for air entering the control room via the main intake (with or without filtration) and also as a result of unfiltered inleakage,

$(\chi/Q)_{intake}$ = atmospheric dispersion factor (sec/m^3), from release point to the main air intake,

$(\chi/Q)_{leakage}$ = atmospheric dispersion factor (sec/m^3), from release point to the MCR envelope (unfiltered inleakage pathway),

F_{intake} = air flow through the main intake (m^3/sec),

$F_{leakage}$ = unfiltered inleakage (m^3/sec), and

f_{bypass} = MCR intake filter bypass fraction, i.e., the fraction of total air intake (sum of main intake flow and unfiltered inleakage) which bypasses the filters. (More unfiltered air enters the MCR atmosphere if the filtration efficiency is less than 100 percent; this is handled automatically in the analytical model.)

Equation 1 is based on RG 1.194, Section 3.3.2.1, Equation 5b, while Equation 2 is from first principles. The equations are based on the fact that the product of intake flow times the atmospheric dispersion factor, e.g., $F_{intake}\ (\chi/Q)_{intake}$, represents that fraction of the atmospheric release of radioactivity, which is about to enter the MCR. The fraction of the atmospheric release which enters the MCR and bypasses the intake filters is:

$$F_{leakage}\ (\chi/Q)_{leakage} = \{ (\chi/Q)_{eff} * [F_{intake} + F_{leakage}] \} * f_{bypass} \quad \text{(Eq. 3)}$$

where the term within the curly brackets is the total fraction of released radioactivity, which is about to enter the control room (prior to filtration), and the second term (i.e., f_{bypass}) is the fraction of the entering radioactivity which bypasses the intake filters.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-7:

The methodology for using the filter bypass fraction in the dose calculations is described in the response to Question 15.00.03-6.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-8:

The main control room (MCR) pressure boundary is located in Safeguard Buildings 2 and 3 on floor Elevation +53 ft and floor Elevation +69 ft. On Elevation +53 ft, it comprises the MCR, technical support center, offices, kitchen, toilet facilities, corridors, and computer room. On Elevation +69 ft, it comprises the MCR air supply, filtration, and air conditioning systems.

No calculations were performed for the direct shine dose to MCR occupants from the radioactive materials in the Reactor Building or the external cloud because of the shielding afforded by the massive concrete structures (≈6 ft).

For the filter shine, the source/receptor geometry is selected so as to maximize the filter shine. The filter is treated as a point source located 6 inches above the ≈20 inch thick floor (at floor Elevation +69 ft) of the upper MCR volume (HVAC location). The dose receptor point is 6 ft above the operating floor (at floor Elevation +53 ft) of the MCR, which is directly below the filter. A total distance of 10.5 ft and a ≈20 inch thick floor (concrete shield) separate the point source and the point receptor.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-9:

The TSC is within the MCR pressure boundary, and the MCR emergency filtration system serves the entire MCR volume.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-10:

The CVCS letdown flow is cooled before it exits the containment for purification. The coolant temperature used in the analysis of the CVCS line break is 122°F. There is no flashing at this temperature.

The bounding small line break scenario is the rupture of a 1/4 inch sampling line connected to the reactor coolant system cross-over leg, with a flow rate of 58,000 lbm/hr and a flashing fraction of 40 percent. This equates to 23,200 lbm/hr of steam flow, which bounds the 17,600 lbm/hr for the CVCS line break scenario with 10 percent flashing.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-11:

The entries in U.S. EPR FSAR, Tier 2, Table 15.0-27 present a condensed set of the time-dependent variables. The flow from the affected steam generator (SG) is terminated 43.8 minutes after event initiation. One minute is assumed for main steam relief isolation valve (MSRIV) closure after reaching the MSRIV low steam pressure setpoint at 42.8 minutes (see U.S. EPR FSAR, Tier 2, Table 15.0-24). The bases for these durations are the sequence of actions provided in U.S. EPR FSAR, Tier 2, Table 15.0-24.

The flow from the intact SGs is conservatively assumed to last for 30 days and represents a bounding scenario.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-12:

No condenser decontamination factor is assumed after reactor trip and LOOP (30 minutes into the event).

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-13:

The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary leakage is 0.125 gpm per intact steam generator (SG) or increases to 0.125 gpm per SG as a result of accident induced conditions. Technical Specifications limiting condition for operation (LCO) 3.4.12.d limits primary-to-secondary leakage through each SG to less than or equal to 150 gallons per day (0.104 gpm), which is less than the assumed analytical value. The assumed value of 0.125 gpm used in the safety analysis conservatively bounds the LCO limit.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-14:

The limiting clad-failure and fuel-melt fractions that yield 90 percent of the dose acceptance criteria at the critical receptor are 3.3 percent and 0.58 percent, respectively. The critical dose receptor location is the main control room in both cases. However, the core thermal-hydraulic analysis of the main steam line break accident shows a clad failure of 1.24 percent without any fuel melt.

FSAR Impact:

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

Question 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).

Response to Question 15.00.03-15:

An average steaming rate of 93.9 lbm/sec was calculated based on the energy dissipated from the time of the postulated main steam line break accident until the introduction of residual heat removal system cooling, approximately 7.26 hours later. The steaming rate was then conservatively increased by 20 percent to 113 lbm/sec, and the cooldown time was conservatively increased from 7.26 to 8 hours. This average steaming flow rate of 113 lbm/sec was held constant using only three SGs (37.7 lbm/sec-SG) for 8 hours.

In the radiological dose calculation, all reactor coolant system leakage into the affected SG is assumed to be released instantly to the atmosphere. Radioactivity entering an unaffected SG is assumed to be released instantly to the atmosphere during tube uncovery (0 to 30 minutes), and is then released to the atmosphere at the SG steaming rate of 37.7 lbm/sec.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-16:

The 0.4 percent value in U.S. EPR FSAR, Tier 2, Section 15.0.3.9.3 and Table 15.0-40 is incorrect. The value used in the dose analysis of the REA is 4 percent. U.S. EPR FSAR, Tier 2, Section 15.0.3.9.3 and Table 15.0-40 will be corrected.

The radiological evaluation of the REA consists of parametric cases combining clad failure and fuel melt assumptions. The acceptable fuel failures are determined on the condition that the total effective dose equivalent dose to any receptor would not exceed 90 percent of the regulatory limit (0.9*6.3 = 5.67 rem) for the off-site receptors and (0.9*5 = 4.5 rem) for the main control room personnel.

The parametric case scenarios involved clad failure only, and both clad failure and fuel melt, the latter being equivalent to 4 percent of the clad failures (i.e., the equivalent on one full rod experiencing complete fuel overheat or melt for every 25 rods that experience clad failure). The 4 percent fuel overheat fraction is used to fix one of the variables in the parametric set of cases. The REA doses reported in U.S. EPR FSAR, Tier 2, Table 15.0-12 correspond to the maximum acceptable clad failure (without fuel melt) equal to 33.4 percent and are dictated by the containment release pathway and the receptor at the exclusion area boundary. The core thermal-hydraulic response calculation does not predict fuel overheat/melt and the maximum clad failure is 26 percent.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.0.3.9.3 and Table 15.0-40 will be revised as described in the response and indicated on the enclosed markup.

Question 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.

Response to Question 15.00.03-17:

The methodology for the rod ejection accident calculation meets the criteria of SRP 4.2, Rev. 3, Appendix B by assuming that the gap fraction for overheated fuel is equal to 100 percent of the fuel rod inventory.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-18:

Aerosol deposition is assumed for the containment leakage pathway, based on natural deposition on containment surfaces. U.S. EPR FSAR, Tier 2, Table 15.0-40 references U.S. EPR FSAR, Tier 2, Table 15.0-41 for the containment leakage pathway.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-19:

A response to this question will be provided by September 19, 2008.

Question 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.

Response to Question 15.00.03-20:

The FHA dose analysis assumes that all of the fuel rods in the dropped assembly fail. This conservative assumption is based on the analysis of the mechanical failure from dropping a similar fuel type, which shows that less than 50 percent of the rods in the dropped assembly would fail (i.e., cladding failure based on exceeding strain limits). Based on engineering judgment, it is concluded that the FHA dose analysis assumption that all rods fail is bounding.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-21:

The equipment hatch and airlocks open inside the Fuel Building, which vents to the main stack, rendering the main stack as the closest release point to the main control room (MCR) air intakes. Potential releases via the material lock (Refer to U.S. EPR FSAR, Tier 2, Figure 2.3-1), if open, would be farther away from the MCR intakes than the main stack.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-22:

Recently irradiated fuel is fuel that has occupied part of the critical reactor core within the previous 34 hours. The definition of “recently irradiated fuel” is in the Technical Specifications 3.9 Bases, “Refueling Operations,” and will be modified for clarification. The analysis of the fuel handling accident (FHA) demonstrates that after the 34 hours of decay time, the maximum off-site total effective dose equivalent resulting from a FHA without credit for containment is 5.62 rem at the exclusion area boundary.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Chapter 16, Technical Specifications 3.9 Bases, “Refueling Operations,” will be revised as described in the response and indicated on the enclosed markup.

Question 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.

Response to Question 15.00.03-23:

A response to this question will be provided by September 19, 2008.

Question 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.

Response to Question 15.00.03-24:

A response to this question will be provided by September 19, 2008.

Question 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.

Response to Question 15.00.03-25:

A response to this question will be provided by September 19, 2008.

Question 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.

Response to Question 15.00.03-26:

A response to this question will be provided by September 19, 2008.

Question 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.

Response to Question 15.00.03-27:

A response to this question will be provided by September 19, 2008.

Question 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.

Response to Question 15.00.03-28

After drawdown of the Shield Building annulus, no mixing of the annulus volume is assumed in the analysis of the loss of coolant accident containment leakage pathway. This assumption meets the guidance of Regulatory Guide 1.183, Appendix A, Section 4.3. For this scenario, containment releases are assumed to be transported directly to the exhaust filtration system and released to the atmosphere.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-29:

The post-loss of coolant accident (LOCA) atmospheric release pathways due to containment leakage are as follows:

- Unfiltered containment purging at the start of the accident, for 10 sec (terminated by automatic containment isolation), at sonic velocity as a result of the post-LOCA pressure transient, via the main stack.
- Unfiltered release of containment leakage at a point adjacent to the steam generator #3 (SG-3) silencer (potential release point closest to the main control room air intakes), during the 305 sec drawdown of the various secondary buildings, without holdup.
- Filtered release (after the end of drawdown) of the containment leakage via the reactor building annulus filtration system (AVS) and the main stack, without holdup.

Containment leakage that bypasses the annulus would enter the Fuel Building and/or the Safeguard Buildings atmospheres and be released to the environment via the safeguard building controlled area ventilation system (CAVS). Because the filtration efficiencies are the same for the AVS and the CAVS, and both systems exhaust to the main stack, the radiological consequences of releases via the CAVS (with no holdup assumed) would be the same as those via the AVS. The releases are modeled as being via the AVS.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-30:

A response to this question will be provided by September 19, 2008.

Question 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.

Response to Question 15.00.03-31:

No mixing of airborne radioactivity (resulting from post-loss of coolant accident engineered safeguard features component leakage) is assumed to occur within the Safeguard Buildings and Fuel Building.

FSAR Impact:

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

Question 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.

Response to Question 15.00.03-32:

The engineered safeguard features (ESF) component leakage release location changes when drawdown of the secondary buildings is complete.

The post-loss of coolant accident atmospheric release pathways, because of ESF component leakage, are as follows:

- Unfiltered release of the airborne source term from ESF component leakage at a point adjacent to the steam generator #3 silencer (i.e., the potential release point closest to the main control room intakes), during the 305 sec drawdown of the various secondary buildings, without holdup.
- Filtered release of the same source term described above at the main stack, after the end of drawdown (filtered by the Safeguard Building controlled area ventilation system), with consideration of the noble gases generated by the iodines retained by the exhaust filter.

FSAR Impact:

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

U.S. EPR Final Safety Analysis Report Markups

alkalis to the RCS for the secondary-side leakage pathway. The release points to the atmosphere are as follows:

- Primary containment leakage pathway: base of main stack.
- Secondary-side leakage pathway: four silencers on top of Safeguard Building Divisions 1 and 4 (with equal steaming rates via each MSRT).

The REA scenario diagrams are shown in Figure 15.0-9—REA - Primary Containment Leakage Scenario Diagram for the primary containment leakage and Figure 15.0-10—REA - Secondary-Side Leakage Scenario Diagram for the secondary-side leakage pathway.

15.0.3.9.3 Results

The potential radiological consequences of an REA are summarized in Table 15.0-44. One of the objectives of the evaluation was to determine the maximum ~~clad~~cladding failure and fuel overheat that can be accommodated for an REA without exceeding 90 percent of the dose acceptance criterion at any receptor. The acceptable fuel failures are as shown in the table, dictated by the dose at the EAB. The overheat fraction is selected to correspond to ~~0.~~4 percent of the ~~clad~~cladding failures and is on a full-core mass basis.

15.00.03-16

15.0.3.10 Fuel Handling Accident

This section addresses the radiological impact associated with a postulated design basis fuel handling accident (FHA) at the U.S. EPR. The accident is postulated to occur either in the Containment Building or in the Fuel Building. The PA scenario is based on the guidance in SRP, Section 15.0.3 (Reference 1) and RG 1.183, Appendix B.

A fuel handling accident is postulated to take place at the start of fuel movement, 34 hrs after reactor shutdown (all rods in). In this PA, it is assumed that the peak-powered assembly, operating at a radial peaking factor of 1.7, drops onto other assemblies. This action leads to fuel damage equivalent to ~~clad~~cladding failure of all 265 fuel rods within the dropped assembly and to the ensuing release of the entire fuel assembly gap inventory.

Other fuel handling accidents, such as a spent fuel cask falling or tipping into the spent fuel pool (SFP), are prevented by the design of the spent fuel handling equipment. The spent fuel cask and transfer machine are located in a separate room from the SFP area, which prevents a cask from being in the SFP area altogether. Fuel handling equipment and procedures are described in Section 9.1.4, and cask handling operations are described in Section 9.1.4.2.1.

Table 15.0-40—Design Input for Rod Ejection Accident
Sheet 1 of 3

Description		Value	References and Remarks
Source Term			
Core inventory		See Table 15.0-14	
Radial peaking factor		1.7	
Fuel rod activity gap fractions	Halogens	10%	
	Noble gases	10%	RG 1.183, Table 3
	Alkalis	12%	
Primary and secondary side coolant radionuclide concentrations		See Tables 15.0-15 & 15.0-16	
Pre-accident halogen spike (assumed to be the same as for the MSLB)		60 µCi/gm DE-I131	RG 1.183, Appendix E, Section 2.1
Reactor Coolant System Variables			
Coolant volume in RCS and pressurizer		15,009 ft3	
Coolant mass in RCS and pressurizer		6.47E+05 lb_m	
Primary to secondary leak rate used in analysis		0.125 gpm/SG	
Secondary Side Coolant Variables			
SG water inventory	100% power	1.698E+05 lb_m/SG	
	Hot shutdown	2.311E+05 lb_m/SG	
	Average	2.005E+05 lb_m/SG	For fractional steaming rate value
Iodine partition coefficient in secondary-side water		100	RG 1.183, Appendix E, Section 5.5.4
Alkali steam carry over fraction		1%	
Primary Containment Leakage Pathway			
Fuel damage (produces approximately 90% of criterion at worst-case receptor)	Cladding failure	Table 15.0-44	Includes rods that overheat/melt
	Full-rod fuel overheat/melt	Table 15.0-44	Set equal to ~~0.~~4% of cladding failure (Source term excludes gap activity)

15.00.03-16

Table 15.0-40—Design Input for Rod Ejection Accident
Sheet 3 of 3

Description		Value	References and Remarks
Exhaust filtration efficiency under accident conditions (PCIS-actuated annulus KLB system)		99%	For 4-inch activated carbon bed and 70% relative humidity
Secondary-Side Leakage Pathway			
Fuel damage (determined to yield approximately 90% of dose acceptance criterion at worst-case receptor)	Clad failure	See Table 15.0-44	Includes rods that overheat
	Full-rod fuel overheat	See Table 15.0-44	Set equal to ~~0.~~4% of clad failure (Source term excludes gap activity)
Fraction of gap activity released to RCS (instantaneous release, uniform mixing)		100%	RG 1.183, Appendix H, Section 1. (The release from overheated fuel is conservatively assumed to be same as that from melted fuel.)
Fraction of overheated fuel inventory released to RCS	Halogens	50%	
	Noble gases	100%	
Plant cooldown average steaming rate via all four MSRTs to RHR cut-in at 250°F RCS temperature)		113 lb_m/s	
Partition coefficient for halogens in SG water		100	RG 1.183, Appendix E, Section 5.5.4
Steam carryover of alkalis		1%	
Chemical composition of halogens released to atmosphere	Elemental	97%	REG 1.183, Appendix H, Section 5
	Organic	3%	
Other Variables			
Time at which plant cooldown is switched from SG steaming to the RHR		8 hrs	
Offsite receptor variables		See Table 15.0-19	
MCR variables		See Table 15.0-18	MCR isolation actuated by PCIS
MCR composite χ/Qs and intake filter bypass fractions for Primary Containment Leakage pathway		See Table 15.0-42	Releases at base of main stack
MCR composite χ/Qs and intake filter bypass fractions for the secondary-side leakage pathway		See Table 15.0-43	Releases via MSRTs/ silencers

15.00.03-16

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

15.00.03-22

During movement of recently irradiated fuel assemblies (i.e., fuel assemblies that have occupied part of a critical core within the previous 34 hours) within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining the containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of Regulatory Guide 1.183, Table 6 (Ref. 1). Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of recently irradiated fuel assemblies within containment, the equipment hatch must be closed and held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operations in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.