AREVAEPRDCPEm Resource

From: Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent: Thursday, January 22, 2009 4:23 PM
To: Getachew Tesfaye
Cc: WILLIFORD Dennis C (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); WELLS Russell D (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 150 (1656, 1606),FSAR Ch. 12
Attachments: RAI 150 Response US EPR DC.pdf

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 150 Response US EPR DC.pdf" provides technically correct and complete responses to 3 of the 8 questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI Questions 12.02-1, 12.02-2 (b), 12.03-12.04-5, and 12.03-12.04-7.

The following table indicates the respective page(s) in the response document, "RAI 150 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 150 — 12.02-1	2	4
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A complete answer is not provided for 5 of the 8 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 150 — 12.02-2(a)	May 14, 2009
RAI 150 — 12.02-3	May 14, 2009
RAI 150 — 12.03-12.04-3 (Part 1)	May 14, 2009
RAI 150 — 12.03-12.04-4	May 14, 2009
RAI 150 — 12.03-12.04-6	May 14, 2009

Sincerely,

Ronda Pederson
ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935

Phone: 434-832-3694
Cell: 434-841-8788

From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Monday, December 08, 2008 3:44 PM
To: ZZ-DL-A-USEPR-DL
Cc: Sara Bernal; Timothy Frye; Tarun Roy; Surinder Arora; Joseph Colaccino; John Rycyna; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 150 (1656, 1606),FSAR Ch. 12

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on December 2, 2008, and on December 8, 2008, you informed us that the RAI is clear and no further clarification is needed. As a result, no change is made to the draft RAI. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs, excluding the time period of **December 20, 2008 thru January 1, 2009, to account for the holiday season** as discussed with AREVA NP Inc. For any RAIs that cannot be answered **within 45 days**, it is expected that a date for receipt of this information will be provided to the staff within the 45-day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 144

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Created By: Ronda.Pederson@areva.com

Recipients:
"WILLIFORD Dennis C (AREVA NP INC)" <Dennis.Williford@areva.com>
Tracking Status: None
"BENNETT Kathy A (OFR) (AREVA NP INC)" <Kathy.Bennett@areva.com>
Tracking Status: None
"DELANO Karen V (AREVA NP INC)" <Karen.Delano@areva.com>
Tracking Status: None
"WELLS Russell D (AREVA NP INC)" <Russell.Wells@areva.com>
Tracking Status: None
"Getachew Tesfaye" <Getachew.Tesfaye@nrc.gov>
Tracking Status: None

Post Office: AUSLYNCMX02.adom.ad.corp

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Response to

Request for Additional Information No. 150 (1656, 1606), Revision 0

12/8/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 12.02 - Radiation Sources
SRP Section: 12.03-12.04 - Radiation Protection Design Features
Application Section: FSAR, Tier 2, Section 11.1, Source Terms; Section 12.1, ALARA; & Section 12.3, Radiation Protection Design Features
Application Section: Section 12.3, Radiation Protection Design Features

QUESTIONS for Health Physics Branch (CHPB)

Question 12.02-1:

RG 1.206, Section C.I.12.2.1, Contained Sources, states, in part, that the applicant should describe the sources of radiation, during normal plant operations and accident conditions, that are the bases for the radiation protection design, such that the facility is demonstrated to be in compliance with 10 CFR Part 20 and GDC 61 in Appendix A of 10 CFR Part 50.

The EPR FSAR, Tier 2, Section 12.1.1.2, ALARA Design and Construction Policies, states that

> “The US EPR design process…is in accordance with the guidance provided in RG 8.8 Section C.2 to achieve occupational doses and doses to members of the public that are ALARA”
>
> Section C.2 of RG 8.8 states that, "to provide a basis for design...fission product source terms should be estimated using a 0.25% fuel cladding defect for PWRs."
>
> In FSAR Tier 2 Section 11.1.2.2, "Design Basis for Shielding," you indicated that you would use this as the basis for the shielding evaluation source term, "except for radiodine, bromines, and noble gases." Instead, you indicated that "radioiodine and bromine are at the TS concentration limits."
>
> The EPR FSAR does not indicate why the technical specification limits were used, nor does it indicate what basis was used to calculate noble gas concentrations. Please explain your rationale and how this is appropriate in light of the guidance provided in section C.2 of RG 8.8. Discuss whether or not the method used is conservative relative to the 0.25% failed fuel fraction assumption described in RG 8.8.

RG 1.206, Section C.I.12.2.1, Contained Sources, states that the applicant should describe the contribution of neutron and gamma streaming to radiation levels in potentially occupied areas of containment. The EPR FSAR, Tier 2, Section 12.3.1.1, Reactor Building, states that personnel will access the service compartments of containment while at power in order to stage equipment for outages. However, the shielding codes referenced in the FSAR solve gamma-ray transport equations only and do not calculate dose rates due to neutron fluxes, therefore there is no neutron dose rate provided in the FSAR for these at-power containment entries.

> In accordance with RG 1.206, describe the contribution of neutron streaming to the radiation levels seen by personnel while accessing the service compartments at power. Describe the method and/or code that was used to estimate the neutron contribution to radiation levels within the service compartment. Relevant operating experience from similar operating reactors may be used.

Response to Question 12.02-1:

Part 1

The radioiodine technical specification for primary coolant is based upon 1.0 µCi/gram dose equivalent (DE) I-131 which in turn is based upon NUREG-1430 and NUREG-1431. This is more conservative than the values calculated using the 0.25 percent failed fuel fraction. The primary coolant radioiodine technical specification value, based upon 1.0 µCi/gram DE I-131, exceeds the calculated value using the bounding core inventory and 0.25 percent failed fuel fraction by a factor of 1.677. This adjustment factor was also applied to I-129 and I-130, which are not included in the DE I-131 definition, and to the bromines.

The radioiodine technical specification for secondary coolant is based upon 0.1 µCi/gram DE I-131 which in turn is based upon NUREG-1430 and NUREG-1431. The secondary radioiodine technical specification concentration exceeds the calculated value by a factor of 157.5. This adjustment factor was also applied to I-129 and I-130, which are not included in the DE I-131 definition, and to the bromines.

The bounding value calculated for noble gases is 210 µCi/gram DE Xe-133. The calculated value is based upon the bounding core inventory (2 to 5 wt% U-235 enrichment, 5 to 62 GWd/MTU burnup, and 0.25 percent failed fuel fraction). This value was subsequently adopted as the technical specification value for noble gases, thus the calculated and technical specification values are identical.

Part 2

The U.S. EPR Reactor Building is a two-compartment design segregating the radioactive reactor coolant system (RCS) components from the personnel access areas. The Monte Carlo code MCNP was used to calculate both gamma and neutron dose rates. All areas and components making up two of the primary loops (approximately half of the Reactor Building) were explicitly modeled within MCNP. The dose rate in the personnel access areas was calculated based on:

- Concrete density of 2.35 g/cc and a polyethylene density of 0.95 g/cc
- Neutron sources streaming through the primary piping openings in the bulk shielding
- Nitrogen-17 sources in the reactor coolant loop.

Section 12.3.2.2 of the U.S. EPR FSAR, Tier 2 will be modified to describe the contribution of neutron streaming to the radiation levels within the service compartments and the computer code and assumptions used for these calculations. Section 12.3.7 will be revised to add the MCNP code reference in accordance with the additional text describing its use.

The shielding thicknesses (walls, doors, ceilings) were designed such that the service compartments are accessible at power. The dose rate criteria used to determine shielding thicknesses are:

- ≤ 2.5 mrem/hr gamma, and
- ≤ 0.1 mrem/hr neutron.

These values were determined at a distance of 1.64 feet from the surface of the component or shielding.

Based on the MCNP results, the maximum dose equivalent rates are in front of the primary piping (around the 16 foot level). At that point, the gamma dose equivalent rate is 2.4 mrem/hr and the neutron dose equivalent rate is 0.04 mrem/hr. Neutron streaming contributes less than 2 percent of the total dose rate.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 12.3.2.2 and Section 12.3.7 will be revised as described in the response and indicated on the enclosed markup.

Question 12.02-2:

Airborne radioactivity sources considered for the design of ventilation systems, radiation monitoring, personal protective measures and for dose assessment should be described by location and magnitude such that there is reasonable assurance that the requirements of 10 CFR 20.1203, 10 CFR 20.1204, and 10 CFR Part 20 Appendix B will be met.

The EPR FSAR, Tier 2, Section 12.2.2, Airborne Radioactive Material Sources, provides a description of the airborne sources located in the plant. However,

a) It is not clear from the parameters listed in Table 12.2-19, Parameters and Assumptions for Calculating Airborne Radioactive Concentrations, how the airborne concentrations were calculated for the EPR. Please provide sample airborne radioactivity concentration calculations for the fuel and reactor buildings such that NRC staff can independently verify the methodology used and the concentrations reported in the FSAR. Include all equations, assumptions and any default values that may have been used.

b) Airborne radioactivity concentrations for Safeguard Buildings 1, 2, 3 and 4 were not provided. Justify the exclusion of these buildings as possible airborne areas or provide estimates of the radioactivity concentrations that could be found in these buildings.

Response to Question 12.02-2:

Part a

Response to be provided by May 14, 2009.

Part b

U.S. EPR FSAR Tier 2, Section 12.3.1.2 describes the facility design features for the Safeguard Building. There are two areas in each division of the Safeguard Building:

> "…the radiological controlled area (consisting of safety injection and vent and drain systems), and the uncontrolled area (containing instrumentation, control equipment, and switchgear). The systems that contain radiation sources are placed closest to the Reactor Building in the bottom two floors. Each of the areas is also served by a separate ventilation system. The ventilation system is divided into two trains of equipment to separately serve the controlled and uncontrolled areas."

Similar to the Nuclear Auxiliary Building and the Radioactive Waste Processing Building, equipment leakage (from the safety injection and vent and drain systems) is the primary source of airborne radioactivity concentrations within the Safeguard Building. The safety injection and vent and drain systems are located in normally unoccupied areas, and are served by separate ventilation systems. Negligible airborne radioactivity concentrations are expected in the Safeguard Building. The Safeguard Building will be added to the discussion of airborne radioactivity concentrations in the U.S. EPR FSAR, Tier 2, Section 12.2.2.1.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 12.2.2.1 will be revised as described in the *Part b)* response and indicated on the enclosed markup.

Question 12.02-3:

Radiation sources that are the basis for the radiation protection program and for shield design calculations should be described such that there is reasonable assurance that the requirements of 10 CFR 20.1201, 20.1206 and 10 CFR 50.34(f)(2)(vii) are met. In addition, RG 1.206, Part C.I.12.2.1, Contained Sources, states, in part,

"The applicant should describe the sources of radiation, during normal plant operations and accident conditions, that are the basis for the radiation protection design."

However, the EPR FSAR in Tier 2, Section 12.2.1, Contained Sources, does not provide information on the following sources which constitute a significant radiation protection and shielding design concern. In accordance with RG 1.206, please provide the following:

i) Spent fuel gamma ray source strengths for maximum burn-up as a function of time after shutdown

ii) Safety Injection System Source strengths at various times following a LOCA.

iii) Normal Residual Heat Removal System source strength as a function of time after shut down.

Response to Question 12.02-3:

Response to be provided by May 14, 2009.

Question 12.03-12.04-3:

1. 10 CFR 20.1501(b) requires licensees to ensure that the instruments and equipment used for quantitative radiation measurements are calibrated periodically for the radiation measured. RG 1.206, Part C.I.12.3.4, Area Radiation and Airborne Radioactivity Monitoring Instrumentation, states that the applicant should provide information regarding the calibration methods and frequency for the monitoring instrumentation.

   The EPR FSAR Tier 1, section 12.3.4, Area and Airborne Radioactivity Monitoring Instrumentation, states that radiation instrumentation complies with 20.1501 and that additional information on calibration of fixed area and airborne radioactivity monitors is provided in the Radiation Protection Program described in Section 12.5. Section 12.5 of the EPR FSAR consists of one COL action item, for which all COL applicant's are referencing NEI 07-03, Generic FSAR Template Guidance for Radiation Protection Program Description. However, NEI 07-03 does not provide any information on the calibration of fixed area and airborne monitors – only calibration of portable monitors.

   In accordance with RG 1.206, provide information on the calibration methods and frequency that will be used for the EPR fixed area and airborne monitors. Discuss to what extent the calibration guidance described in ANSI/ANS 6.8.1, Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors will be incorporated. If not, describe the specific alternative approaches used.
2. EPR FSAR, Tier 2, Figure 12.3-24, Safeguard Buildings 2 and 3 -31 Ft Elevation Radiation Zones, shows 4 pumps labeled as "Medium Head Safety Injection System Pumps," although there are two Low Head Safety Injection System Heat Exchangers located in elevation -16 Ft of the same buildings, as shown on Figure 12.3-25, Safeguard Buildings 2 and 3 -16 Ft Elevation Radiation Zones. Correct this typo, or provide clarification.
3. In Tier 2, EPR FSAR, Section 12.3.2.2, Shielding Calculation Methods, the following statement is made: "selection of buildup factors [is]based on the last significant shield material the radiation passes through." Provide more detail as to how the "last significant shield material" was selected, such that NRC staff can reproduce the shielding calculation methodology described in this section of the FSAR.

Response to Question 12.03-12.04-3:

Part 1

Response to be provided by May 14, 2009.

Part 2

The typographical error on U.S. EPR FSAR Tier 2, Figure 12.3-24 will be corrected to reflect that equipment labeled C and D in the "Equipment Legend" of this figure are "Low Head Safety Injection System Pump" as opposed to "Medium Head Safety Injection System Pump".

Part 3

The last significant shield material is the last thick material through which the radiation will travel. The most conservative approach is to select the material that results in the greatest buildup. As a general rule, this is either the last shield between the source and the dose point or the most dominant shield (i.e., the shield with the most "mean free paths"). Typically, in the majority of the cases analyzed, the concrete of the wall of the room under consideration is selected for the buildup contribution to the dose rate results.

FSAR Impact:

U.S. EPR FSAR Tier 2, Figure 12.3-24 will be revised as described in *Part 2* of the response and will be included in the update of radiation zone figures in response to RAI 43, Supplement 2, Question 14.03.08-1 (scheduled to be submitted to the NRC by 2/27/2009).

Question 12.03-12.04-4:

Facility design should minimize the potential for creating very high radiation areas during normal operations, including abnormal operational occurrences. High and very high radiation areas should be isolated from normally occupied rooms and corridors such that personnel access to these areas can be controlled in accordance with 10 CFR 20.1601, 10 CFR 20.1602 and the guidance in Regulatory Guide 8.38.

Section 4.4.6.2 of the EPR FSAR, Tier 2 describes the aeroball system as "an electromechanical, computer-controlled, online flux mapping measurement system based on movable activation probes (aeroballs)." These "aeroballs" can be moved into and out of the core through the top of the reactor vessel using a pneumatic transport system. In currently operating reactors movable in-core detectors have become stuck during transit outside of the reactor vessel, thereby creating high or very high radiation areas that have resulted in unplanned personnel exposures. For the aeroball system, it is not clear from the description provided whether the activation probes could become stuck in transit outside of the core and result in unplanned worker exposures, particularly during outages.

i) Discuss what design features and access controls will be in place to ensure that if one or more aeroballs become stuck outside of the core (in transit to the measurement room, for example) workers occupying containment during outages would not receive radiation doses in excess of 10 CFR Part 20 occupational dose limits. Discuss how doses to workers performing maintenance on the aeroball system, including the activation probes themselves, will be maintained below 10 CFR Part 20 limits. Discuss how control of the radioactive aeroballs will be maintained if the balls had to be replaced, particularly since the aeroballs are so small in diameter (0.067 inches) and there are so many of them (2500 aeroballs per stack, 40 stacks in a core). If the aeroball system is in use at currently operating plants, describe any operating experience that may justify the use of such a system.

ii) Provide information or a drawing describing where and how the activation probes are stored following transit from the core (and assuming they are not located in the measurement room). Provide information or a drawing describing the shielding at the storage location to ensure that dose rates to personnel in the area are maintained ALARA.

iii) Provide the isotopic composition and source strengths for the aeroballs following the maximum expected use time at 100% power (including stuck detectors). Also provide source strengths projected over a 3 week period (such as a refueling outage).

iv) EPR FSAR, Tier 2, Figure 4.4-11, Aeroball Probe, indicates that a carrier gas will be used to transport the activation probes. This gas will be located in the core at times and will therefore become activated. Provide information on what gas will be used in the system, how much activity (and what isotopes) will be contained in the gas following the maximum expected in-core time at 100% power, as well as how the used radioactive gas will be disposed of. Discuss what design features will be in place to maintain occupational doses due to the carrier gas ALARA.

Response to Question 12.03-12.04-4:

Response to be provided by May 14, 2009.

Question 12.03-12.04-5:

The area radiation monitoring system must meet the provisions of 10 CFR 20.1501, 10 CFR 50.34(f)(2)(xxvii) and GDC 63.

RG 1.206, Part C.I.12.3.4, Area Radiation and Airborne Radioactivity Monitoring Instrumentation, states that the applicant should describe the criteria for selection and placement of the fixed radiation monitors in accordance with ANSI/ANS-HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation monitoring Systems for Light-Water Nuclear Reactors." ANSI/ANS-6.8.1, Section 4.2, Detector Locations, has the following criteria for locating instrumentation:

"… 4.2.2 Special consideration should be given to those normally accessible and occasionally accessible areas which can experience significantly greater exposure rates resulting from operational transients or maintenance activities."

Examples of areas provided in ANSI/ANS 6.8.1 that may meet the criteria described above include:

- the radwaste building, including solid radwaste storage areas, the drumming station control panel area and the radwaste control panel areas
- the primary sample station area,
- hot machine shops,
- decontamination areas,
- the reactor water cleanup heat exchanger area,
- the residual heat removal pump and heat exchanger areas,

EPR FSAR Section 12.3.4.1.1, Normal Operations, states that EPR instrumentation selection and placement follow the criteria of ANSI/ANS-6.8.1. EPR FSAR Tier 2, Table 12.3-3, Radiation Monitor Detector Parameters, describes the locations of fixed area radiation monitors for the EPR design. However, Table 12.3-3 of the EPR FSAR does not list any area monitors for the radwaste building, contrary to the guidance provided in ANSI/ANS 6.8.1, and contrary to the commitment provided in EPR FSAR Section 11.4.1.2.4, Controlled Releases, which states:

"…area radiation monitors throughout the Radioactive Waste Processing Building detect excessive radiation levels and alert the operators to this condition, in accordance with GDC 63. Area radiation monitoring is addressed in detail in Section 12.3.4."

1. Provide the locations of area radiation monitors in Table 12.3-3 for the radioactive waste processing building in accordance with the guidance provided in ANSI/ANS 6.8.1, the commitment provided in Section 11.4 of the EPR FSAR, and the requirements of GDC 63.

2. For several EPR areas which are encompassed by the bulleted list above (including the primary sampling room, the radiochemistry laboratory, the hot workshop located in the nuclear auxiliary building, the CVCS heat exchanger area, and the residual heat removal pump and heat exchanger areas located in the safeguard buildings), no area monitors

are provided in Table 12.3-3. Modify Table 12.3-3 to include area monitors at these locations or discuss how workers will be alerted of increasing radiation levels in these and other areas where radiation levels could increase significantly due to transients or maintenance, and where personnel may be present.

Response to Question 12.03-12.04-5:

Part 1

The design includes two area radiation monitors in the Radioactive Waste Processing Building (RWB). They are located in the decontamination room and the drumming room. These area radiation monitors for the RWB were added to U.S. EPR FSAR Tier 2, Table 12.3-3 as part of the response to RAI 105, Question 11.04-4.

Part 2

The design includes area radiation monitors in the Nuclear Auxiliary Building (NAB) for the protection of personnel. These monitors are located in the primary sampling, active laboratory, filter replacement machine (already listed in the U.S. EPR FSAR), and hot workshop rooms. Additions to the U.S. EPR FSAR Tier 2, Table 12.3-3 are presented in Table 12.03-12.04-5-1.

Table 12.03-12.04-5-1—Radiation Monitors in NAB

Monitor Location	Monitor Provisions		Range
	Continuous	ACF	
NAB	1 monitor – in the primary sampling room (elevation -31')	---	1E-5 – 1E+0 rem/hr
	1 monitor – in the active laboratory (elevation -31')	---	1E-5 – 1E+0 rem/hr
	1 monitor – in the hot workshop (elevation +64')	---	1E-5 – 1E+0 rem/hr

Note: The chemical and volume control system (CVCS) heat exchangers are located in a locked, inaccessible area of the Reactor Building during normal operation in which monitoring for personnel safety is not required.

FSAR Impact:

U.S. EPR FSAR Tier 2, Table 12.3-3 will be revised as described in the response to *Part 2* and indicated on the enclosed markup.

Question 12.03-12.04-6:

10 CFR 20.1601 requires that lockable doors be provided to all high radiation areas. The EPR FSAR, Tier 2, Section 12.3.1.8.2, Fuel Building, states that

"Occupied areas adjacent to the fuel transfer tube are shielded so that dose rates are less than 100 rads per hour during fuel movement operations, in accordance with section 12 of the NUREG-0800."

Figures 12.3-9, "Containment Building Section Looking Plant-East as the Reactor Cavity, Core Internals Storage, Transfer pit, and Spreading Area," and Figure 12.3-4, "Transfer pit at the +17 Ft Elevation in the Reactor Building," provide a view of the spent fuel transfer pit as well as areas adjacent to the pit within the reactor building that could be occupied. Provide information on the minimum concrete thickness between the pit and these potentially occupied areas within the reactor building, as well as the estimated dose rates in these areas during spent fuel transfer. Discuss design features that will be in place to control access to high radiation areas adjacent to the fuel transfer pit during fuel transfer (for example, the "access to transfer pit" rooms shown in Figures 12.3-34, "Fuel Building +12 Ft Elevation Radiation Zones," and 12.3-4).

Response to Question 12.03-12.04-6:

Response to be provided by May 14, 2009.

Question 12.03-12.04-7:

EPR Tier 2, Section 12.3.4.1.3, In-containment High-Range Monitoring, states that the in-containment monitoring instrumentation used during postulated accidents meet the requirements of IEEE Std 497-2002 for a Type C, category 3 instrument and a Type E, category 1 instrument. However, IEEE Std 497-2002 does not discuss “categories.” This standard only groups instrumentation into Types A, B, C, D, or E. Please correct this statement in the FSAR.

Response to Question 12.03-12.04-7:

The U.S. EPR FSAR statement will be corrected to delete the word “categories”.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 12.3.4.1.3 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

12.2.1.12 Post-LOCA ESF Filters

The radiation shielding source terms for the ESF filters post-LOCA are listed in Table 12.2-18—Photon Spectra for ESF Filters Post-LOCA.

12.2.1.13 Miscellaneous Sources

A combined license (COL) applicant that references the U.S. EPR design certification will provide site-specific information for required radiation sources containing byproduct, source, and special nuclear material that may warrant shielding design considerations. This site-specific information will include a listing of isotope, quantity, form, and use of all sources in this latter category that exceed 100 millicuries.

12.2.2 Airborne Radioactive Material Sources

Airborne radioactive material sources in the plant are considered in the design of the ventilation systems. Airborne radioactivity is monitored inside the plant, as described in Section 12.3.4, and in process equipment and effluents, as described in Section 11.5.

12.2.2.1 Normal Operations

Airborne radioactivity concentrations can occur in the Reactor Building, both during power operation (coolant leakage) and refueling (evaporation of the refueling pool). The normal airborne radioactivity concentrations within the Containment Building are based on continuous RCS leakage into the equipment area of the Reactor Building and activation of naturally-occurring argon in the Reactor Building air that is exposed to high neutron flux level, with subsequent leakage to the service area. The assumptions and parameters listed in Table 12.2-19—Parameters and Assumptions for Calculating Airborne Radioactive Concentrations were used to evaluate the airborne radionuclide concentrations in the Reactor Building.

The spent fuel pool water contains radionuclides from defects in spent fuel and corrosion products released from fuel assemblies. The evaporation of the spent fuel pool water then leads to airborne radioactivity concentrations in the Fuel Building, both during power operation and refueling. The airborne radioactivity in the Fuel Building is primarily because of tritium, since the continuous operation of the FPCPS effectively removes other isotopes from the pool. The assumptions and parameters listed in Table 12.2-19 were used to evaluate the airborne radionuclide concentrations in the Fuel Building. The concentrations in the Reactor and Fuel Buildings are listed in Table 12.2-20—Airborne Radioactivity Concentrations.

Equipment leakage is the primary source of airborne radioactivity concentrations within the Safeguard Building, Nuclear Auxiliary Building and Radioactive Waste Processing Building. This equipment is located in normally unoccupied areas. The ventilation systems in the Safeguard Building, Nuclear Auxiliary Building and

12.02-2 (b)

Radioactive Waste Processing Building are designed so that the airflow is from regions of lower potential for contamination to those with higher potential for contamination, and then exhausted from the building. As a result, negligible airborne radioactivity concentrations are expected in those areas of the Safeguard Building, Nuclear Auxiliary Building and Radioactive Waste Processing Buildings which are normally occupied.

12.02-2 (b)

As discussed in Section 12.2.1.6, components within the Turbine Building are considered to be nonradioactive under normal operating conditions (no primary-to-secondary leaks). Thus, airborne radioactivity concentrations in the Turbine Building are also expected to be negligible.

12.2.3 References

1. NUREG-0800, "U.S. NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 2007.

2. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors PWR-GALE Code," Revision 1, U.S. Nuclear Regulatory Commission, April 1985.

12.02-1 (2)

The RANKERN computer code (Reference 5) was used to determine the dose rates within the lower elevations of Safeguard Building Divisions 1 and 2 for post-LOCA conditions. The RANKERN code uses a Monte Carlo treatment of the source and is in common usage in Europe.

The gamma and neutron dose rates in the Reactor Building were calculated using MCNP, Version 4c (Reference 10). Within the Reactor Building, all areas and components making up two primary loops (approximately half of the entire building) were modeled. The dose rate in the personnel access areas was calculated based on:

- Concrete density of 2.35 g/cc and a polyethylene density of 0.95 g/cc.
- Neutron sources streaming through the primary piping openings in the bulk shielding.
- Nitrogen-17 sources in the reactor coolant loop.

The radiation sources listed below identify the location of major equipment and how doses to personnel are minimized.

Reactor Vessel

The reactor vessel is located low in the center of the Reactor Building and is well shielded. The shielding arrangement is shown in Figures 12.3-7 and 12.3-8.

Reactor Coolant System

Each RCP and each SG are located in individual compartments in the equipment compartment of the Reactor Building, providing shielding from each other as well as the service compartment of the Reactor Building (see Section 12.3.1.1). The shielding that separates the equipment compartment from the service compartment provides sufficient shielding to enable personnel to enter the Reactor Building during power operation (see Figure 12.3-13).

Chemical and Volume Control System

The CVCS components and piping are located in the Reactor Building and the Fuel Building. On the letdown portion of the system, the regenerative heat exchanger (elevation +5 feet) and the high pressure (HP) coolers (elevation -8 feet) are each located in separate well-shielded compartments of the Reactor Building. The compartments for these components are designed with removable shield walls.

Shielded pipe ducts are provided for the letdown piping from the Reactor Building to the Fuel Building as well as within the Fuel Building. The volume control tank, which is located in a compartment by itself, spans elevations +0 feet and +12 feet of the Fuel Building. A shielded anteroom provides access to the tank room at elevation +12 feet

- Record data from the monitors to maintain a record of the gamma radiation after an accident as a function of time so that the inventory of radioactive materials in the containment volume can be estimated.

- Initiate Reactor Building containment ventilation system isolation and exhaust filtering on high radiation inside the Reactor Building.

- Initiate Safeguard Building controlled-area ventilation system isolation and exhaust filtering on high radiation inside the Reactor Building.

- Initiate Fuel Building ventilation system exhaust filtering on high radiation in the Reactor Building, concurrent with Safeguard Building controlled area ventilation system alignment.

12.03-12.04-7

These safety-related instruments and the associated network are environmentally qualified (refer to Section 3.11) to survive an accident and perform their design functions. The instruments are designed to respond to gamma radiation over the energy range of at least 60 keV to 3 MeV, with a dose rate response accuracy within a factor of two over the entire range. These monitors conform to the criteria set forth in 10 CFR 50.34(f)(2)(xvii), NUREG-0737, II.F.1 (Reference 3), and RG 1.97 (refer to Section 7.5). These monitors also meet the requirements of IEEE Std 497-2002 for a Type C~~, category 3~~ instrument and a Type E~~, category 1~~ instrument (Reference 7). This instrumentation is powered by the EUPS (refer to Section 8.3.1), which is served by a two-hour battery backup with diesel generators as the auxiliary power to provide continuous indication.

The in-containment high-range monitoring instrumentation consists of four independent high-range monitors located in widely separated areas in the service compartment of the containment. The locations are chosen to allow the detectors to be exposed to a significant volume of the containment atmosphere without obstruction so that the readouts are representative of the containment atmosphere, yet permitting easy access for calibration and maintenance activities.

Table 12.3-3 includes the high-range monitoring instrumentation.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

12.3.4.2.1 Normal Operations

The airborne radioactivity monitoring instrumentation for use during normal operation and AOOs is provided to:

- Continuously monitor for the presence of airborne radioactivity at selected locations of the plant that are normally occupied and may contain airborne radioactivity.

12.3.6.4 Minimization of Radioactive Wastes

Waste is minimized by reducing the source of waste through design features that limit contamination. This design philosophy minimizes waste activity and volume both during plant operation and ultimate decommissioning.

12.3.7 References

1. NUREG-0800, "U.S. NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission~~NRC~~, March 2007.

2. ANSI/ANS-6.4-1997, "Specification for Radiation Shielding Materials," American National Standards Institute, 1997.

3. NUREG-0737, "Clarifications of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission~~NRC~~, November 1980.

4. "MicroShield® User's Manual," Version 7, Grove Software, Inc., Lynchburg, VA, October 2006.

5. "RANKERN Version 15a – A Point Kernel Integration Code for Complicated Geometry Problems," Serco Assurance, October 2005.

6. ANSI/ANS-HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation. Monitoring Systems for Light Water Nuclear Reactors," American National Standards Institute/American Nuclear Society, May 1981.

7. IEEE Standard 497-2002, "Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Inc, 2002.

8. ANSI/HPS-N13.1-1999, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities," American National Standards Institute, 1999.

9. NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities," U.S. Nuclear Regulatory Commission~~NRC~~, December 1997.

10. LA-13709-M, "MCNP–A General Monte Carlo N-Particle Transport Code," Version 4c, J.F. Briesmeister, Ed., Los Alamos National Laboratory, April 2000.

12.02-1 (2)

Table 12.3-3—Radiation Monitor Detector Parameters
Sheet 2 of 3

Monitor Location	Monitor Provisions: Continuous	Monitor Provisions: ACF	
	2 monitors in air leaving containment – next to air duct (downstream KLA2 low flow purge exhaust)	---	1E-5 – 1E+(
	2 monitors in exhaust air from exhaust cell (downstream KLB accident exhaust filter)	---	1E-4 – 1E+4 (Kr-85, Xe-
	2 monitors in exhaust air from exhaust cell (downstream KLC accident exhaust filter)	---	1E-4 – 1E+4 (Kr-85, Xe-
Safeguard Building (Mechanical)	1 monitor – Service Corridor near Containment Heat Removal System (elevation -31' Division 1)	---	1E-4 – 1E+4
	1 monitor – Service Corridor near Safety Injection System (elevation -31' Division 2)	---	1E-4 – 1E+4
	1 monitor – Service Corridor near Safety Injection System (elevation -31' Division 3)	---	1E-4 – 1E+4
	1 monitor – Service Corridor near Containment Heat Removal System (elevation -31' Division 4)	---	1E-4 – 1E+4
	1 monitor – Personnel Air Lock Area (elevation 0' Division 2)	---	1E-4 – E+4
Safeguard Building (Electrical)	1 monitor – MCR (+53' elevation Division 2/3)	---	1E-4 – 1E+4
Nuclear Auxiliary Building	1 monitor – Filter Changing Equipment Room (elevation 0')	---	1E-4 – 1E+1
	1 monitor – In the primary sampling room (elevation -31')	---	1E5 – 1E+0

12.03-12.04-5 (2)

12.03-12.04-5 (2)

**Table 12.3-3—Radiation Monitor Detector Parameters
Sheet 3 of 3**

Monitor Location	Monitor Provisions: Continuous	Monitor Provisions: ACF	
	1 monitor – In the active laboratory (elevation -31')	---	1E5 – 1E+0
	1 monitor – In the hot workshop (elevation +64')	---	1E5 – 1E+0
Post Accident Monitoring			
Reactor Building	4 monitors inside containment – Service Compartment	Signals Reactor Building air filtration isolation and RHR valve closure	1E-1 – 1E+7
Radioactive Waste Processing Building	1 monitor – In the drumming room next to conveyor	---	1E-4 – 1E+4
	1 monitor – In the decontamination room	---	1E-4 – 1E+4