ArevaEPRDCPEm Resource

From:	Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent:	Monday, July 13, 2009 3:51 PM
To:	Tesfaye, Getachew
Cc:	BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); GUCWA Len T (EXT)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 235, FSAR Ch. 12
Attachments:	RAI 235 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 235 Response US EPR DC.pdf," provides technically correct and complete responses to 2 of the 2 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 235 Question 12.03-12.04-11.

The following table indicates the respective pages in the response document, "RAI 235 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 235 — 12.02-4	2	3
RAI 235 — 12.03-12.04-11	4	6

This concludes the formal AREVA NP response to RAI 235, and there are no questions from this RAI for which AREVA NP has not provided responses.

Sincerely,

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From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Friday, June 12, 2009 11:13 AM
To: ZZ-DL-A-USEPR-DL
Cc: Bernal, Sara; Frye, Timothy; Jennings, Jason; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 235 (2851, 2850),FSAR Ch. 12

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on May 19, 2009, and discussed with your staff on June 3, 2009. Draft RAI Question 12.03-12.04-11 was modified as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 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: 648

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Response to

Request for Additional Information No. 235 (2851, 2850), Revision 0

6/12/2009

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 Ch. 12

QUESTIONS for Health Physics Branch (CHPB)

Question 12.02-4:

According to RG 1.206, Section C.I.12.2.2, Airborne Radioactive Material Sources, the applicant should provide a description of airborne radioactive material sources in the plant considered in the design of the ventilation systems and used for the design of personnel protective measures as well as for dose assessment.

In the EPR FSAR Section12.2, Table 12.2-19, "Parameters and Assumptions for Calculating Airborne Radioactive Concentrations" a reactor coolant tritium concentration of 1 uCi/g for calculating airborne radionuclide concentrations is listed, presumably based on the guidance provided by ANSI/ANS-18.1-1999, "Radioactive source term for normal operation of light water reactors." However, ANSI/ANS-18.1 - 1999 is based on reactor coolant tritium data for a PWR reference plant with a 12 month refueling cycle, making it non-conservative for all currently operating plants which follow a 24 month refueling cycle, and making it particularly non-conservative for those plants which are designed to recover boron and recycle tritiated water, as is the case with the US EPR (See section 9.3.4, "Chemical and Volume Control System (Boron Recovery System)." SRP Section 12.2 states that for PWRs designed to recycle tritiated water, tritium concentrations in contained sources as well as tritium airborne concentrations should be based on a primary coolant concentration of 3.5 uCi/g. Provide revised airborne tritium concentrations for the Nuclear Island based on a primary coolant tritium source term of 3.5 uCi/g or higher, or justify the use of the 1 uCi/g concentration.

Response to Question 12.02-4:

The airborne tritium concentrations in the Reactor Building and Fuel Building are as follows:

Reactor Building

The ANSI/ANS 18.1-1999 tritium concentration is a function of the inventory of tritiated liquids in the plant, the rate of production of tritium due to activation in the reactor coolant, the rate of release from the fuel, and the extent to which water is recycled or discharged from the plant. The tritium concentration given in ANSI/ANS 18.1-1999 Tables 6 and 7 is representative of PWRs with a moderate amount of tritium recycle.

U.S. EPR FSAR, Tier 2, Table 11.1-2, shows that a reactor coolant tritium concentration of 1.0 μCi/gm is the nominal or "expected" concentration during normal reactor operation. This value is based on data from several multi-year tritium following programs carried out in pressurized water reactors with Zircaloy clad fuel. The value of 1.0 μCi/gm is estimated from the measured coolant activities and represents a realistic value for the expected concentration in the reactor coolant system. Those programs also showed that 4.0 μCi/gm is the estimated bounding or design value for the tritium concentration.

The U.S. EPR Reactor Building airborne tritium concentration is directly proportional to the reactor coolant tritium concentration. The tritium airborne radioactivity concentration in the service area of the Reactor Building is 9.26E-07 μCi/ml (see U.S. EPR FSAR, Tier 2, Table 12.2-20). This value is based on coolant leakage into the Reactor Building from the reactor coolant system that contains the nominal tritium concentration of 1.0 μCi/gm identified above.

U.S. EPR FSAR, Tier 2, Table 11.1-2, shows that 4.0 μCi/gm is the maximum design basis tritium concentration in reactor coolant. Because the airborne tritium concentration is directly proportional to the reactor coolant tritium concentration, for the design basis coolant concentration the airborne tritium concentration in the Reactor Building service area will increase by a factor of four to 3.70E-06 µCi /ml (see U.S. EPR FSAR, Tier 2, Table 12.2-20). This concentration is well below the 10 CFR Part 20 Appendix B inhalation derived air concentration value of 2E-05 µCi/ml.

Fuel Building

Airborne activity in the Fuel Building is due to evaporation from the spent fuel pool water. The tritium activity in the spent fuel pool is due to the release of tritium through fuel cladding defects in the stored fuel assemblies. Mixing of the reactor coolant water and spent fuel pool water is not a significant source of tritium activity in the spent fuel pool. Thus, the airborne tritium activity in the Fuel Building as shown in U.S. EPR FSAR, Tier 2, Table 12.2-20 is independent of the reactor coolant tritium concentration.

The tritium activity in the spent fuel pool water is calculated based on a full core off-load following three years of power operation, assuming that tritium from the spent fuel diffuses into the spent fuel pool water at the rate of one percent of the total core inventory per year.

FSAR Impact:

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

Question 12.03-12.04-11:

EPR FSAR Tier 2, Section 12.3.5.2, “Post accident Access to Radiological Vital Areas,” lists the following radiological vital areas:

1. MCR, technical support center, and adjoining rooms
2. Safeguard Building containment heat removal system pump rooms
3. Safeguard Building residual heat removal system pump rooms,
4. Post-LOCA sampling room in the Fuel Building
5. Post-LOCA ventilation air sampling room in the Fuel Building
6. Radiological analysis laboratory in the Nuclear Auxiliary Building
7. Diesel fuel oil delivery area.

Section 12.3.5.2 states that for the above listed radiological vital areas, mission doses were calculated to be less than 5 rem total effective dose equivalent (TEDE), in accordance with 10 CFR 50.34(f)(2)(vii) and GDC 19, and in accordance with NUREG-0737 II.B.2. The FSAR also notes that for missions (2), (3), (4), (5), (6) and (7) listed above, "the operator wears full protective clothing and respiratory protection, therefore only direct dose is considered."

In response to RAI 136 Question 12.03-12.04-2 part (3) the applicant stated that the COL holder's radiation protection program would ensure that doses remained ALARA during post-accident radiological vital area access, including ensuring the appropriate use of respiratory protection and temporary shielding, as necessary.

However, while the use of respiratory protection can limit intakes significantly, its ability to reduce intake doses to negligible levels will be dependent on the post-accident airborne concentrations present in the vital areas at the time that access is required.

Therefore, demonstrate the U.S. EPR design’s compliance with GDC 19 and 10 CFR 50.34(f)(2)(vii) by providing airborne concentrations for all vital areas or by providing information (such as additional mission description detail and/or additional design detail) that would support the conclusion that post-LOCA doses due to airborne radioactivity (from sources such as ESF system leakage or stack releases) would be negligible for missions (2) through (7). Update the FSAR to include this information.

Response to Question 12.03-12.04-11:

The radiological vital area doses for missions (2) through (6) have been revised to include the external dose from immersion in the airborne activity, as well as direct shine from contained sources. Because mission dose (7) occurs outside plant structures, the immersion dose was already included in the dose for this mission. The potential inhalation dose is excluded for missions (2) through (6) because credit is taken for respiratory protection. Although respiratory protection is also planned for mission (7), the calculated dose rate includes the inhalation contribution.

The airborne concentrations presented in U.S. EPR FSAR, Tier 2, Table 12.03-13 represent the highest airborne concentrations in the radiological vital areas (given entry at the earliest expected time or later). The concentrations are time-dependent, and the maximum values presented correspond to the time when the inhalation dose would have been the greatest if no credit had been taken for respiratory protection. Therefore, these radiological concentrations constitute the maximum challenge to the respiratory protection credit.

The airborne concentrations consider both:

- Leakage from the Containment Building before the Shield Building annulus depressurization is complete (i.e., annulus bypass during annulus drawdown)
- The filtered release from the plant stack after the necessary negative pressure in the Shield Building annulus has been established and the Containment Building leakage is being collected.

The combination of the airborne activity concentration from these two Containment Building leakage sources (annulus bypass leakage prior to 305 seconds post- loss of coolant accident (post-LOCA) and the plant stack release after 305 seconds post-LOCA) constitute the basis for the activity concentrations presented in U.S. EPR FSAR, Tier 2, Table 12.03-13.

Any annulus bypass leakage entering the Fuel Building and Safeguard Buildings (where radiological vital area access may be needed) would be expected to first enter penetration areas where personnel access is not required. Further, the operation of the safety-related ventilation system would draw this leakage away from the radiological vital areas requiring access. Notwithstanding this expected behavior, to simplify the analysis, the annulus bypass leakage that occurs prior to complete drawdown of the Shield Building annulus is conservatively assumed to enter and mix with the atmosphere of either the Fuel Building alone (for missions (4) and (5)) or the combined Fuel and Safeguard Buildings (for missions (2) and (3)) assuming, in both cases, 50 percent mixing with the available volume.

Radioactive leakage from the Containment Building after the Shield Building annulus drawdown is complete is assumed to be collected, filtered, and released from the plant stack. It is then assumed to be drawn into the Fuel Building and Safeguard Buildings ventilation intake (after credit for atmospheric dispersion using the same methodology as that used to calculate atmospheric dispersion for the Main Control Room) and assumed to also mix with 50 percent of the Fuel Building atmosphere (for missions (4) and (5)) or the combined Fuel and Safeguard Buildings atmosphere (for missions (2) and (3)).

The treatment of annulus bypass leakage described above is applied only for assessing missions (2), (3), (4), and (5). For the Nuclear Auxiliary Building (mission (6)), there is no pathway for the annulus bypass leakage to enter the structure directly, so the bypass is conservatively assumed to be released directly to the environment without hold-up or filtration (as is also the case for mission (7)). This assumption is consistent with the expectation that annulus bypass leakage during the first 305 seconds post-LOCA, if it occurred, would be collected in penetration areas and immediately released; but it is also conservative in that no credit is taken for the filtration of the collected bypass during that period.

Because annulus bypass leakage cannot directly enter the Nuclear Auxiliary Building, contamination of the Nuclear Auxiliary Building atmosphere occurs only as the result of contaminated outside air entering the structure. Therefore, while the Nuclear Auxiliary Building ventilation is not safety-related, it is conservatively assumed to continue to run. As with the Fuel Building and Safeguard Buildings, atmospheric dispersion between the plant stack and the Nuclear Auxiliary Building air intake is calculated using the same methodology as that used to calculate atmospheric dispersion for the Main Control Room.

To maintain analytical consistency with the activity concentrations presented in U.S. EPR FSAR, Tier 2, Table 12.03-13, the radiological vital area access immersion doses due to the airborne activity have been added to the radiation doses from contained sources previously calculated in accordance with NUREG-0737 Item II.B.2. The radiological vital area access mission doses presented in U.S. EPR FSAR, Tier 2, Table 12.3-12 will be revised to include this additional dose contribution.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 12.3.5.2 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

- Figures 12.3-64 through 12.3-71 contain postaccident radiation zone maps that encompass the identified radiological vital areas. The radiation zones for Division 4 of the Safeguard Building are the same as those for Division 1 (symmetrical layout). These zones were determined in conformance with the source term assumptions of RG 1.183.

- For higher dose rate areas, actions such as flushing of pumps and lines and installation of local temporary shielding are used to reduce dose rate in area to 100 mrem/hr. Thus, a higher dose rate is used during preparatory work, and a lower dose rate is used after shielding installation and flushing operations are complete.

- Occupancy values used for the MCR, technical support center, and adjoining rooms are in accordance with RG 1.183.

Additional mission specific assumptions are as follows:

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- MCR, technical support center, and adjoining rooms. The two sources of radiation are airborne radioactivity (because of containment and ESF leakage resulting in both an immersion and inhalation dose) and direct radiation from the intake filters and from the recirculation filters located in the floor above the MCR. External shine from the Reactor Building and annulus structure is not a significant contributor to the dose rate because of the presence of substantial concrete shielding.

- Containment heat removal system or residual heat removal systems to clear blockage, back flushing of sump screens. The operator wears full protective clothing and respiratory protection, thus only direct dose is considered. Access begins no sooner than 20 hours post-LOCA.

- Post-LOCA grab samples. The operator wears full protective clothing and respiratory protection, thus only direct dose is considered. The sample lines within the sample room are the only significant sources of exposure. The first samples are drawn no sooner than 13 hours post-LOCA and are then transported to the laboratory in a shielded container.

- Post-LOCA ventilation air samples. The operator wears full protective clothing and respiratory protection, thus only direct dose is considered. The samples are transported in a shielded container to the laboratory.

- Process samples in the laboratory. The operator wears full protective clothing and respiratory protection, thus only direct dose is considered. Temporary shielding will be used as necessary so that the sampling box is the only significant source of exposure. The degassing vessel is the primary source of exposure within the sampling box.

- Diesel fuel delivery. The operator wears respiratory protection during delivery, thus only direct dose is considered.

Access routes for each radiological vital area within buildings are shown in Figures 12.3-75 through 12.3-79. For the diesel fuel delivery, trucks enter via the

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security access facility and proceed to the fill valve located on the outside of the Emergency Power Generating Buildings 1 and 4. Table — summarizes each radiological vital area mission, including the dose rate, mission time (time to access the area, perform the task, and exit the area), and total mission dose.

Table 12.3-13—Maximum Airborne Concentrations in Radiological Vital Areas at or Beyond the Time of First Access presents the highest airborne concentrations in each of the radiological vital areas requiring post-accident access. These concentrations are based on post-accident Containment Building leakage and are consistent with entry into each of the radiological vital areas at the earliest expected time or later. The concentrations are time-dependent, and the maximum values presented correspond to the point in time when the inhalation dose is the greatest if no credit is taken for respiratory protection. Therefore, these concentrations constitute the maximum challenge to the respiratory protection credit.

The maximum external exposure doses for immersion for collection of post-accident ventilation air samples and sample analysis (included in the doses presented in Table 12.3-12) correspond to different airborne activity concentrations from those presented in Table 12.3-13. This is because the greatest potential for inhalation dose (averted by the use of respiratory protection) for those missions does not occur at the same time as the maximum external exposure dose. Other areas of the Fuel Building and Safeguard Buildings (e.g., Shield Building penetration areas) may exhibit higher activity concentrations, but those areas do not require post-accident access.

12.3.5.3 Dose to the Public from Direct Radiation Exposure at the Exclusion Area Boundary

The annual radiation dose at the exclusion area boundary of 0.5 mi because of direct radiation from the Containment Building, Fuel Building, and other contained radioactive sources within the U.S. EPR plant site is less than 1 mrem and meets the limits of 10 CFR 20.1301(e) and 40 CFR 190. The dose rate from direct radiation at the site boundary does not exceed 2 mrem in any one hour in accordance with 10 CFR 20.1301(a)(2). The U.S. EPR design also provides storage for refueling water (IRWST) inside the Containment Building, instead of in an outside storage tank, thereby eliminating the refueling water storage tank as an offsite radiation source.

12.3.6 Minimization of Contamination

The U.S. EPR design complies with the requirements of 10 CFR 20.1406 by applying a contaminant management philosophy to the design of structures, systems, and components (SSC), which have the potential to contain radioactive materials. The principles embodied in this philosophy are prevention of unintended release, and early detection, if there is unintended release of radioactive contamination. The application of the contaminant management philosophy leads to design features that maintain

Table 12.3-12—U.S. EPR Estimated Accident Mission Dose
Sheet 1 of 2

Mission	Dose per Person	Area Dose Rate	Occupancy Time
Staffing of MCR, TSC, nearby stations			
Contained Sources (filter shine)	0.1 rem	Varies	30 day
Airborne Radioactivity (immersion/inhalation)	3.9 rem	Varies	30 day
Total	4.0 rem	N/A	30 day
Access to SAHRS System			
Contained Sources	2.7 rem	0.2 rem/hr 0.7 rem/hr 0.1 rem/hr	0.5 hr access/return 2 hr preparatory work 12 hr repair
Airborne Radioactivity (immersion)	0.7 rem	0.047 rem/hr	14.5 hr (0.5 + 2 + 12)
Total	3.4 rem	N/A	14.5 hr
Access to RHR System			
Contained Sources	2.3 rem	0.3 rem/hr 0.4 rem/hr 0.1 rem/hr	1 hr access/return 2 hr preparatory work 12 hr repair
Airborne Radioactivity (immersion)	0.7 rem	0.047 rem/hr	15 hr (1 + 2 + 12)
Total	3.0 rem	N/A	15 hr
Post-Accident Sampling (IRWST Liquid, Containment Atmosphere)			
Contained Sources	2.3 rem (1.05 rem extremity)	8.84 rem/hr (63 rem/hr extremity) 0.1 rem/hr	0.25 hr in area 1 min obtain sample 0.5 hr transport route
Airborne Radioactivity (immersion)	0.06 rem	0.08 rem/hr	0.77 hr (0.25 + 1/60 + 0.5)
Total	2.4 rem	N/A	0.77 hr
Post-Accident Sampling (Ventilation Air Sample)			
Contained Sources	0.38 rem (obtain samples) (<1 mrem extremity) <1 mrem (transport)	2.3 rem/hr (4.7 rem/hr extremity) 2.5 mrem/hr	10 min in area to obtain sample 0.22 hr transport route (access/return)
Airborne Radioactivity (immersion)	0.06 rem	0.16 rem/hr	0.39 hr (10/60 + 0.22)
Total	0.45 rem	N/A	0.39 hr

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Table 12.3-12—U.S. EPR Estimated Accident Mission Dose
Sheet 2 of 2

Mission	Dose per Person	Area Dose Rate	Occupancy Time
Sample Counting Lab			
Contained Sources	1.0 rem	9.2 rem/hr (adjacent to sampling box) 100 mrem/hr (low dose-rate area)	10 min in area (about 1/3rd of time in low dose-rate area where operator moves during processing)
Airborne Radioactivity (immersion)	0.09 rem	0.52 rem/hr	10 min
Total	1.1 rem	N/A	10 min
Diesel Fuel Oil delivery (per delivery)			
Airborne Radioactivity (immersion/inhalation)	0.5 rem	0.5 rem/hr	1 hr
~~Staffing of MCR, TSC, nearby stations~~	~~4.0 rem (3.9 rem from immersion/inhalation 0.1 rem from direct shine from filters)~~	~~Varies~~	~~30 day~~
~~Access to SAHRS system~~	~~2.7 rem~~	~~200 mrem/h~~ ~~700 mrem/h~~ ~~100 mrem/h~~	~~0.5 h access/return~~ ~~2 hr preparatory work~~ ~~12 h repair~~
~~Access to RHR system~~	~~2.3 rem~~	~~300 mrem/h~~ ~~400 mrem/h~~ ~~100 mrem/h~~	~~1.0 h access/return~~ ~~2 hr preparatory work~~ ~~12 h repair~~
~~Postaccident sampling (RCS Liquid, Containment Atmosphere)~~	~~2.3 rem (1.05 rem extremity)~~	~~8.84 rem/h (63 rem/h extremity)~~ ~~100 mrem/h~~	~~0.25 h in area, including 1 minute obtain sample~~ ~~0.5 h transport route~~
~~Postaccident sampling (ventilation air samples)~~	~~383 mrem (obtain samples)~~ ~~< 1 mrem (extremity)~~ ~~< 1 mrem (transport)~~	~~2.3 rem/h (4.7 mrem/h extremity)~~ ~~2.5 mrem/h~~	~~10 minutes in area to obtain sample~~ ~~0.22 h transport route (access/return)~~
~~Sample counting lab (operator moves to lower dose rate area during processing)~~	~~1.0 rem~~	~~9.2 rem/h (adjacent to sampling box)~~ ~~100 mrem/hr (low dose-rate area)~~	~~10 minutes in area (about 1/3rd of time in low dose-rate area)~~
~~Diesel Fuel Oil Delivery (per delivery)~~	~~0.5 rem~~	~~500 mrem/h~~	~~1 hour~~

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Table 12.3-13—Maximum Airborne Concentrations in Radiological Vital Areas at or Beyond the Time of First Access
Sheet 1 of 3

Nuclide	Fuel Building at 1 hr post-LOCA (μCi/cc)	Safeguards Buildings at 20 hrs post-LOCA (μCi/cc)	Auxiliary Building at 1.8 hrs post-LOCA (μCi/cc)	Diesel Fuel Delivery Area at 4 days Post-LOCA (μCi/cc)
Kr-83m	7.43E-05	1.29E-05	1.82E-03	0.00E+00
Kr-85m	1.71E-04	2.45E-04	4.40E-03	1.33E-09
Kr-85	9.24E-06	2.51E-04	2.70E-04	1.74E-04
Kr-87	2.30E-04	1.99E-07	4.35E-03	0.00E+00
Kr-88	4.41E-04	1.16E-04	1.06E-02	0.00E+00
Kr-89	1.42E-09	0.00E+00	1.15E-12	0.00E+00
Xe-131m	6.77E-06	1.79E-04	1.98E-04	2.19E-04
Xe-133m	3.90E-05	8.93E-04	1.14E-03	5.84E-04
Xe-133	1.27E-03	3.22E-02	3.71E-02	2.20E-02
Xe-135m	1.25E-04	3.67E-04	5.46E-03	9.93E-05
Xe-135	4.35E-04	6.01E-03	1.27E-02	1.12E-04
Xe-137	2.13E-08	0.00E+00	1.05E-10	0.00E+00
Xe-138	5.73E-05	0.00E+00	1.60E-04	0.00E+00
Br-83	2.45E-05	9.49E-09	5.85E-06	0.00E+00
Br-84	1.64E-05	0.00E+00	1.73E-06	0.00E+00
Br-85	3.72E-11	0.00E+00	0.00E+00	0.00E+00
I-129	1.39E-11	1.33E-12	4.19E-12	1.93E-12
I-130	2.09E-05	6.88E-07	6.00E-06	1.41E-08
I-131	2.31E-04	2.07E-05	6.94E-05	2.29E-05
I-132	2.49E-04	2.59E-06	6.39E-05	2.53E-06
I-133	4.69E-04	2.38E-05	1.37E-04	2.75E-06
I-134	2.42E-04	8.80E-12	4.30E-05	0.00E+00
I-135	4.06E-04	5.30E-06	1.12E-04	2.67E-09
Rb-86	9.67E-07	7.30E-08	2.18E-07	8.79E-10
Rb-88	2.95E-04	1.19E-04	1.15E-04	0.00E+00
Rb-89	2.75E-05	0.00E+00	8.44E-07	0.00E+00
Cs-134	1.08E-04	8.39E-06	2.44E-05	1.13E-07
Cs-136	2.68E-05	2.00E-06	6.04E-06	2.29E-08
Cs-137	4.12E-05	3.21E-06	9.30E-06	4.34E-08

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Table 12.3-13—Maximum Airborne Concentrations in Radiological Vital Areas at or Beyond the Time of First Access
Sheet 2 of 3

Nuclide	Fuel Building at 1 hr post-LOCA (μCi/cc)	Safeguards Buildings at 20 hrs post-LOCA (μCi/cc)	Auxiliary Building at 1.8 hrs post-LOCA (μCi/cc)	Diesel Fuel Delivery Area at 4 days Post-LOCA (μCi/cc)
Cs-138	2.29E-04	0.00E+00	3.24E-05	0.00E+00
Sr-89	1.39E-07	9.47E-07	4.08E-06	1.81E-08
Sr-90	6.98E-09	9.98E-08	4.26E-07	1.99E-09
Sr-91	7.95E-08	2.84E-07	4.58E-06	2.21E-11
Sr-92	6.84E-08	7.59E-09	3.41E-06	0.00E+00
Y-90	1.48E-10	2.03E-08	1.27E-08	1.29E-09
Y-91m	2.72E-08	1.81E-07	2.18E-06	1.41E-11
Y-91	8.34E-10	1.77E-08	5.25E-08	3.79E-10
Y-92	1.45E-08	5.77E-08	1.31E-06	0.00E+00
Y-93	9.02E-10	3.50E-09	5.22E-08	0.00E+00
Zr-95	9.45E-10	1.34E-08	5.77E-08	2.58E-10
Zr-97	9.63E-10	6.32E-09	5.69E-08	5.58E-12
Nb-95	9.46E-10	1.35E-08	5.78E-08	2.69E-10
Mo-99	1.32E-08	1.55E-07	8.02E-07	1.39E-09
Tc-99m	1.17E-08	1.48E-07	7.15E-07	1.34E-09
Ru-103	1.25E-08	1.76E-07	7.62E-07	3.32E-09
Ru-105	8.66E-09	6.37E-09	4.67E-07	0.00E+00
Ru-106	7.39E-09	1.06E-07	4.51E-07	2.09E-09
Rh-103m	1.13E-08	1.59E-07	6.87E-07	2.99E-09
Rh-105	9.05E-09	1.01E-07	5.52E-07	4.56E-10
Rh-106	7.39E-09	1.06E-07	4.51E-07	2.09E-09
Sb-125	3.97E-09	5.67E-08	2.42E-07	1.13E-09
Sb-127	1.85E-08	2.29E-07	1.12E-06	2.58E-09
Sb-129	4.27E-08	2.90E-08	2.30E-06	0.00E+00
Te-127m	2.52E-09	3.60E-08	1.54E-07	7.14E-10
Te-127	1.85E-08	2.49E-07	1.13E-06	3.17E-09
Te-129m	7.31E-09	1.03E-07	4.46E-07	1.93E-09
Te-129	4.73E-08	1.02E-07	2.71E-06	1.26E-09
Te-131m	2.06E-08	1.90E-07	1.23E-06	6.54E-10

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Table 12.3-13—Maximum Airborne Concentrations in Radiological Vital Areas at or Beyond the Time of First Access Sheet 3 of 3

Nuclide	Fuel Building at 1 hr post-LOCA (μCi/cc)	Safeguards Buildings at 20 hrs post-LOCA (μCi/cc)	Auxiliary Building at 1.8 hrs post-LOCA (μCi/cc)	Diesel Fuel Delivery Area at 4 days Post-LOCA (μCi/cc)
Te-131	2.80E-08	4.27E-08	6.55E-07	1.47E-10
Te-132	2.03E-07	2.45E-06	1.23E-05	2.49E-08
Te-134	9.55E-08	0.00E+00	2.63E-06	0.00E+00
Ba-137m	3.90E-05	3.03E-06	8.80E-06	4.11E-08
Ba-139	6.55E-08	6.63E-11	2.67E-06	0.00E+00
Ba-140	1.04E-07	1.42E-06	6.33E-06	2.39E-08
La-140	2.81E-09	4.35E-07	2.56E-07	2.10E-08
La-141	8.35E-10	4.19E-10	4.43E-08	0.00E+00
La-142	6.20E-10	1.76E-12	2.64E-08	0.00E+00
Ce-141	2.31E-09	3.26E-08	1.41E-07	6.07E-10
Ce-143	2.31E-09	2.21E-08	1.38E-07	8.94E-11
Ce-144	1.76E-09	2.51E-08	1.07E-07	4.96E-10
Pr-143	9.37E-10	1.39E-08	5.73E-08	2.69E-10
Pr-144	1.66E-09	2.51E-08	1.06E-07	4.96E-10
Nd-147	3.89E-10	5.30E-09	2.37E-08	8.67E-11
Np-239	3.90E-08	4.42E-07	2.36E-06	3.47E-09
Pu-238	1.51E-11	2.16E-10	9.21E-10	4.30E-12
Pu-239	6.34E-13	9.10E-12	3.87E-11	0.00E+00
Pu-240	1.45E-12	2.07E-11	8.83E-11	0.00E+00
Pu-241	2.62E-10	3.75E-09	1.60E-08	7.48E-11
Am-241	1.19E-13	1.72E-12	7.27E-12	0.00E+00
Cm-242	5.41E-11	7.71E-10	3.30E-09	1.52E-11
Cm-244	2.87E-11	4.11E-10	1.75E-09	8.19E-12

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