ArevaEPRDCPEm Resource

From: Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent: Wednesday, May 13, 2009 6:35 PM
To: Getachew Tesfaye
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); WILLIFORD Dennis C (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 150, FSAR Ch. 12 Supplement 1
Attachments: RAI 150 Supplement 1 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 3 of the 8 questions of RAI No. 150 on January 22, 2009. The attached file, “RAI 150 Supplement 1 Response US EPR DC.pdf” provides technically correct and complete responses to the remaining 5 questions, as committed.

The following table indicates the respective pages in the response document, “RAI 150 Supplement 1 Response US EPR DC.pdf ,” that contain AREVA NP’s response to the subject questions.

Question #	**Start Page**	**End Page**
RAI 150 —12.02-2(a)	2	8
RAI 150 —12.02-3	9	18
RAI 150 — 12.03-12.04-3 (Part 1)	19	19
RAI 150 — 12.03-12.04-4	20	33
RAI 150 — 12.03-12.04-6	34	40

This concludes the formal AREVA NP response to RAI 150, and there are no questions from this RAI for which AREVA NP has not provided responses.

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: Pederson Ronda M (AREVA NP INC)
Sent: Thursday, January 22, 2009 4:17 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

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
RAI 150 — 12.02-2	5	5
RAI 150 — 12.02-3	6	6
RAI 150 — 12.03-12.04-3	7	8
RAI 150 — 12.03-12.04-4	9	9
RAI 150 — 12.03-12.04-5	10	11
RAI 150 — 12.03-12.04-6	12	12
RAI 150 — 12.03-12.04-7	13	13

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: 478

Mail Envelope Properties (5CEC4184E98FFE49A383961FAD402D31E6E3EE)

Subject: Response to U.S. EPR Design Certification Application RAI No. 150, FSAR Ch. 12 Supplement 1
Sent Date: 5/13/2009 6:35:13 PM
Received Date: 5/13/2009 6:35:31 PM
From: Pederson Ronda M (AREVA NP INC)

Created By: Ronda.Pederson@areva.com

Recipients:
"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
"WILLIFORD Dennis C (AREVA NP INC)" <Dennis.Williford@areva.com>
Tracking Status: None
"Getachew Tesfaye" <Getachew.Tesfaye@nrc.gov>
Tracking Status: None

Post Office: AUSLYNCMX02.adom.ad.corp

Files	**Size**	**Date & Time**
MESSAGE	4534	5/13/2009 6:35:31 PM
RAI 150 Supplement 1 Response US EPR DC.pdf		3584548

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

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 12.02-2:

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.

Response to Question 12.02-2:

Part a)

The airborne concentrations were calculated for U.S. EPR fuel and reactor buildings using the ELISA-2 computer code considering leakage from the reactor coolant system (RCS) using guidance in RG 1.206, Section C.I.12.2.2. The production of Ar-41 was separately determined.

Reactor Building

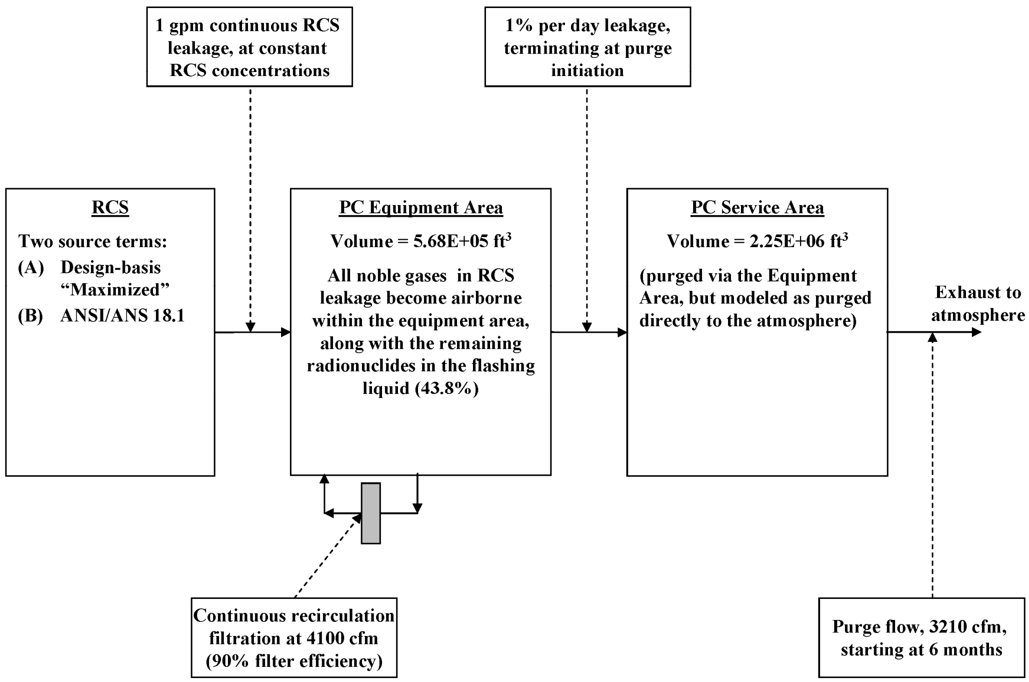
The following two containment representations were used in the analysis:

a) Continuous air mixing between the equipment and service areas (such that there is a uniform concentration of airborne radioactivity within the entire primary containment), modeled as a single compartment.

b) No air mixing between the equipment and service areas, modeled as two separate compartments; one representing the equipment area where the RCS leakage takes place, and the other representing the service area, getting contaminated as a result of gaseous leakage from the equipment area.

Values for the parameters shown in U.S. EPR FSAR Tier 2, Table 12.2-19 were input into ELISA-2 computer code to model the releases in the Reactor Building as shown in Figure 12.02-2-1.

Figure 12.02-2-1—Diagram for Primary Confinement Service Area Airborne Activity During Normal Operation



The resulting airborne concentrations are shown in U.S. EPR FSAR Tier 2, Table 12.2-20 ("Reactor Building Concentration – Service Area" column).

Reactor Building Ar-41 Production

The production of Ar-41 results from the neutron activation of Ar-40 in the reactor pit air space (i.e., the space between the reactor vessel and the concrete biological shield). During normal operation, the air in the reactor pit is cooled by the containment building ventilation system (CBVS). The supply and exhaust ducts that extend outside the equipment area are airtight and maintain isolation of the equipment area from other areas in the Reactor Building. There are four 50 percent supply fans to the reactor pit area, and it is conservatively assumed that mixing the air in the reactor pit with that in the equipment area is instantaneous with no credit for radioactive decay between the pit area and the equipment area. The Ar-41 concentration in the atmosphere of the equipment area is thus equal to the total Ar-41 produced in the reactor pit area divided by the free volume of the equipment area.

The production of Ar-41 in the reactor pit is given by the expression:

$$\frac{dN}{dt} = \Sigma(E)\,\varphi(E,i) - \lambda N,\ atoms/cc - \sec$$

where,

N = number of Ar-41 atoms in the reactor pit atmosphere, atoms/cc

Σ(E) = macroscopic absorption cross section of Ar-40 at neutron energy E, cm^{-1}

φ(E, i) = neutron flux of energy E at point i in the reactor pit, n/cm^2-sec

Solving for the activity, the equation becomes:

$$\lambda N = \Sigma(E)\,\varphi(E,i)\,(1 - e^{-\lambda t}),\ dis/cc - \sec$$

It can be reasonably assumed that Ar-40 is a 1/v absorber, hence the product of Σ(E) φ(E, i) then becomes Σ_a φ(i) where Σ_a is the thermal neutron absorption cross-section and φ(i) is the thermal neutron flux at point i. Since Ar-41 has a relatively short half-life of 1.83 hours, it reaches equilibrium in a relatively short time and the equilibrium activity is simply Σ_a φ (r,θ,z). Since φ(i) $\equiv$ φ(r,θ,z), the total Ar-41 produced in the reactor pit area is then given by:

$$(Ar - 41\,Activity) = \Sigma_a \iiint \varphi(r,\theta,z)\,dr\,d\theta\,dz,\ dis/\sec$$

where,

r = radial distance from vessel to concrete

θ = azimuthal angle around the reactor vessel

z = axial distance in reactor pit

Two-dimensional neutron fluxes in the reactor cavity during full power operation have been calculated with the DORT code for both r-θ and r-z coordinate frames. The r-θ integrated flux is the total flux in the reactor pit over a unit (1 cm) height whose source is an infinite core in the z direction. The total integrated flux in the reactor pit, φ_t, is obtained by multiplying the r-θ integrated flux by the z dimension over which the total flux is desired. This assumes a constant flux in the z dimension which is approximately equal to the flux in the reactor pit averaged over the height of the core.

The integrated flux per unit height, full core symmetry, $\iint \varphi$(r,θ,z) dr dθ, is 1.352E+14 n/sec. The active core height is 420 cm, therefore for this height, the total integrated flux in the reactor pit, $\iiint \varphi$(r,θ,z) dr dθ dz, is 5.678E+16 n-cm/sec. Accounting for the Ar-41 activation above and below the active core height, it was conservatively assumed that the average flux extends 90 cm above and below the active core. This extends the flux by a factor of 1.429. By applying this factor to the flux, the total integrated flux in the reactor pit in which Ar-40 activation to Ar-41 occurs is 8.114E+16 n-cm/sec.

For a 1/v absorber, the macroscopic cross-section is the following:

$$\Sigma_a = \sigma_a N$$

where.

σ_a = microscopic thermal absorption cross-section, cm^2

N = Number density of the target material (e.g., Ar-40 in the Reactor Pit), atoms/cm^3

At a nominal air temperature of 130°F in the reactor pit, the macroscopic cross-section is the following:

$$\Sigma_a = \sigma_a N = 1.098E\text{-}07\ cm^{-1}$$

The product of Σ_a and $\iiint \varphi(r,\theta,z)\, dr\, d\theta\, dz$ is 8.909E+09 dis/sec (2.408E+05 µCi).

The equipment area free volume is 16,079 m^3. Thus the activity, which is the total Ar-41 produced in the reactor pit divided by the equipment area free volume is 1.497E-05 µCi/cc.

If continuous mixing is assumed (i.e., no hold-up or decay between sub-compartments) between the equipment and service areas, the dilution volume of the combined areas is 79,840 m^3 and the specific Ar-41 is 3.016E-06 µCi/cc.

Fuel Building

The radioactivity inventory within the spent fuel pool water is introduced by the following mechanisms:

1) Mixing of the reactor coolant water and spent fuel pool water during refueling operations (this source is considered to be of minimal significance).

2) Release of radionuclides from defects in spent fuel.

3) Release of corrosion products from fuel assemblies due to movement of fuel or normal water circulation/chemistry.

To determine the spent fuel pool water activity, the bounding case of a full core (241 assemblies) off-load was considered. The release of activity from failed fuel is represented by the escape rate coefficients for fuel within the reactor core. However, fuel within the spent fuel pool is at a lower temperature and hence, the escape rate coefficients were modified. *Water Coolant Technology of Power Reactors* by Paul Cohen (American Nuclear Society, 1980) showed a difference of 1E-3 in the escape rate coefficients for I-131 from fuel within the core to fuel outside of the core. Using this information, multipliers used in ELISA-2 to convert the source from Ci to µCi/sec (except for tritium) were calculated as follows:

Multiplier = (failed fuel fraction)*(escape rate coefficient)*(cold fuel factor)*(conversion factor)

= (0.0025) * (escape rate coefficient) * (1E-03) * (1E+06 µCi/Ci)

The multiplier used for tritium was based on a 1 percent diffusion rate from the tritium design basis activity calculation:

H^3 Multiplier = (0.01) (1E+06 μCi/Ci) / (3.15E+07 sec/yr) = 3.17E-04

To determine the ELISA-2 source term value for activation products, the values (μCi/g) were multiplied by a factor of 80 to arrive at the typical Co-60 water concentrations listed in IAEA Technical Report Series No. 218, "Storage of Water Reactor Spent Fuel in Water Pools – Survey of World Experience" (1982) for U.S. PWR Spent Fuel Pools utilizing filters & ion exchange (Trojan, Surry 1-2, Prairie Island). This factor of 80 incorporates the volume of the spent fuel pool and the amount of activation products released from the spent fuel itself.

The airborne activity in the Fuel Building is primarily a result of evaporation from the spent fuel pool. The following equation was used to calculate the evaporation rate from reservoirs of water:

$$E = 0.771(1.465 - 0.0186\ B)\ (0.44 + 0.118\ W)\ (es - ed)$$

where,

E = Evaporation (inches per day)

B = barometric reading (inches of Hg)

W = water surface wind (miles per hour)

es = mean vapor pressure at temperature of water surface (inches of Hg)

ed = mean vapor pressure of saturated air (inches of Hg)

For the Fuel Building calculations, a barometric pressure of 29.5 inches Hg, a wind velocity of 0.02 mph (accounting for area ventilation), a vapor pressure of 5.9 inches Hg for the water surface of the spent fuel pool, and a vapor pressure range from 0.11 to 1.21 inches Hg for the air in the Fuel Building were used. These calculations were based on a Fuel Building temperature ≤ 96 ºF, a Fuel Building relative humidity of 70 percent, and a spent fuel pool water temperature of 140ºF. An evaporation rate between 1.47 and 1.81 inches per day (or 0.79 to 0.98 gpm) was calculated. To establish the airborne activity in the Fuel Building, a conservative value of 1 gpm was used.

The spent fuel pool water activities and Fuel Building airborne activities are listed in Table 12.02-2-1.

The key input parameters for ELISA-2 are contained in U.S. EPR FSAR Tier 2, Table 12.2-19. Input to the ELISA-2 code was prepared using the same methodology as that used for the Reactor Building, as previously described. The results are presented in U.S. EPR FSAR Tier 2, Table 12.2-20 ("Fuel Building Concentration" column).

Table 12.02-2-1—Water and Airborne Activity in the Fuel Building

(Page 1 of 2)

Nuclide	SFP Water Activity ($\mu Ci/cm^3$)	FAB Airborne Activity ($\mu Ci/cm^3$)
H 3	5.90E-01	2.67E-06
Na 24	1.13E-06	5.10E-12
Cr 51	6.01E-06	2.72E-11
Mn 54	3.40E-06	1.54E-11
Fe 55	2.57E-06	1.17E-11
Fe 59	6.06E-07	2.75E-12
Co 58	9.50E-06	4.30E-11
Co 60	1.14E-06	5.15E-12
Zn 65	1.08E-06	4.91E-12
Br 83	2.43E-17	1.07E-22
Kr 83m	4.87E-16	2.13E-21
Kr 85m	1.18E-10	5.28E-16
Kr 85	4.33E-03	1.96E-08
Kr 87	3.71E-26	1.60E-31
Kr 88	6.73E-14	2.98E-19
Rb 88	6.79E-14	3.06E-19
Sr 89	1.52E-07	6.89E-13
Sr 90	1.69E-08	7.67E-14
Sr 91	1.42E-10	6.39E-16
Sr 92	4.58E-18	2.02E-23
Y 90	4.69E-09	2.13E-14
Y 91m	7.84E-11	3.56E-16
Y 91	3.01E-08	1.36E-13
Y 92	8.42E-16	3.75E-21
Y 93	3.95E-11	1.78E-16
Zr 95	3.50E-08	1.59E-13
Nb 95	3.67E-08	1.66E-13
Mo 99	1.75E-05	7.92E-11
Tc 99m	9.60E-06	4.38E-11
Ru 103	3.59E-08	1.63E-13
Ru 106	2.27E-08	1.03E-13
Rh 103m	3.24E-08	1.47E-13
Rh 106	2.27E-08	1.03E-13
Ag 110m	3.83E-10	1.74E-15
Te 127m	2.41E-07	1.09E-12
Te 129m	6.52E-07	2.95E-12
Te 129	4.24E-07	1.92E-12
Te 131m	1.92E-07	8.70E-13
Te 131	4.33E-08	1.96E-13

Table 12.02-2-1—Water and Airborne Activity in the Fuel Building

(Page 2 of 2)

Nuclide	SFP Water Activity (μCi/cm^3)	FAB Airborne Activity (μCi/cm^3)
Te 132	7.92E-06	3.59E-11
Te 134	2.09E-45	8.64E-51
I 129	1.08E-11	4.87E-17
I 130	6.09E-08	2.74E-13
I 131	1.26E-04	5.72E-10
I 132	2.87E-05	1.27E-10
I 133	1.29E-05	5.83E-11
I 134	1.67E-35	7.03E-41
I 135	1.11E-08	4.96E-14
Xe 131m	2.81E-04	1.27E-09
Xe 133m	1.39E-04	6.30E-10
Xe 133	1.67E-02	7.54E-08
Xe 135m	3.39E-09	1.37E-14
Xe 135	4.68E-06	2.11E-11
Cs 134	1.64E-04	7.44E-10
Cs 136	3.16E-05	1.43E-10
Cs 137	6.29E-05	2.85E-10
Ba 137m	5.92E-05	2.69E-10
Ba 140	2.00E-07	9.04E-13
La 140	6.93E-08	3.15E-13
Ce 141	3.29E-08	1.49E-13
Ce 143	4.26E-09	1.93E-14
Ce 144	2.70E-08	1.22E-13
Pr 143	3.19E-08	1.44E-13
Pr 144	2.70E-08	1.22E-13
W 187	3.17E-07	1.43E-12
Np 239	1.73E-07	7.81E-13

FSAR Impact:

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

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:

Part i)

The U.S. EPR spent fuel gamma ray source strengths, as a function of time after shutdown, are presented in Table 12.02-3-1. The spectra were based on the radionuclide mix containing the bounding core inventory for each individual radionuclide of importance for the following core parameters (as listed in U.S. EPR FSAR Tier 2, Table 15.0-14):

Power level:	4612 MW_t
U-235 fuel enrichment:	2 to 5 $^{w}/_{o}$
Fuel burn-up:	5 to 62 GWD/MTU
Core loading:	129 MTU (metric tons of uranium)

The 18-group spectral structure corresponds to that of ORIGEN-2, and the tabulation includes 18 decay times covering the interval from 5 minutes to approximately 23 years.

Table 12.02-3-1—U.S. EPR Full-Core Bounding Photon Spectra at 4612 MW_t
(Page 1 of 3)

Photon Energy (MeV)	Photon Spectra (MeV/sec) as a Function of Shutdown Time (hrs)					
	0.0833 (5 min)	1	2	5	10	20
0.01	1.76E+18	1.44E+18	1.36E+18	1.30E+18	1.22E+18	1.08E+18
0.025	4.15E+17	3.22E+17	2.92E+17	2.59E+17	2.34E+17	2.03E+17
0.0375	1.11E+18	7.58E+17	6.93E+17	6.49E+17	6.15E+17	5.61E+17
0.0575	5.26E+17	4.76E+17	4.71E+17	4.57E+17	4.36E+17	3.98E+17
0.085	7.54E+18	3.65E+18	2.81E+18	2.52E+18	2.36E+18	2.12E+18
0.125	1.16E+19	1.10E+19	1.07E+19	1.03E+19	9.70E+18	8.70E+18
0.225	2.24E+19	1.73E+19	1.62E+19	1.50E+19	1.39E+19	1.19E+19
0.375	2.19E+19	9.79E+18	7.41E+18	6.00E+18	5.36E+18	4.75E+18
0.575	6.41E+19	4.64E+19	4.11E+19	3.49E+19	3.05E+19	2.54E+19
0.85	1.06E+20	6.95E+19	5.66E+19	4.28E+19	3.71E+19	3.24E+19
1.25	8.23E+19	4.35E+19	3.18E+19	2.04E+19	1.39E+19	9.21E+18
1.75	4.5E+19	2.94E+19	2.50E+19	2.01E+19	1.76E+19	1.59E+19
2.25	3.13E+19	1.59E+19	1.06E+19	5.12E+18	2.73E+18	1.90E+18
2.75	1.45E+19	6.98E+18	4.25E+18	1.70E+18	9.75E+17	8.64E+17
3.5	7.09E+18	2.24E+18	1.27E+18	3.27E+17	4.87E+16	1.05E+16
5	2.28E+18	4.07E+16	2.71E+16	1.20E+16	3.52E+15	3.07E+14
7	3.77E+10	3.77E+10	3.77E+10	3.77E+10	3.77E+10	3.77E+10
9.5	5.88E+09	5.88E+09	5.88E+09	5.88E+09	5.88E+09	5.88E+09
Total	4.20E+20	2.59E+20	2.11E+20	1.62E+20	1.37E+20	1.15E+20

Table 12.02-3-1—U.S. EPR Full-Core Bounding Photon Spectra at 4612 MW_t

(Page 2 of 3)

Photon Energy (MeV)	Photon Spectra (MeV/sec) as a Function of Shutdown Time (hrs)					
	50 (2.08 days)	100 (4.17 days)	200 (8.33 days)	500 (20.8 days)	1000 (41.7 days)	2000 (83.3 days)
0.01	7.64E+17	4.34E+17	1.52E+17	1.89E+16	5.35E+15	2.53E+15
0.025	1.55E+17	1.14E+17	7.08E+16	3.10E+16	1.54E+16	6.90E+15
0.0375	4.41E+17	3.23E+17	2.11E+17	1.05E+17	5.99E+16	3.62E+16
0.0575	3.09E+17	2.14E+17	1.17E+17	3.11E+16	6.45E+15	1.98E+15
0.085	1.58E+18	1.00E+18	4.51E+17	9.84E+16	2.91E+16	1.42E+16
0.125	6.31E+18	3.81E+18	1.63E+18	5.40E+17	3.36E+17	1.81E+17
0.225	7.94E+18	4.45E+18	1.59E+18	1.91E+17	4.37E+16	1.27E+16
0.375	3.77E+18	2.80E+18	1.80E+18	7.12E+17	2.11E+17	4.61E+16
0.575	1.85E+19	1.44E+19	1.12E+19	7.83E+18	5.81E+18	4.16E+18
0.85	2.66E+19	2.26E+19	1.90E+19	1.50E+19	1.22E+19	8.77E+18
1.25	5.77E+18	4.19E+18	2.81E+18	1.47E+18	8.13E+17	4.76E+17
1.75	1.46E+19	1.34E+19	1.09E+19	5.58E+18	1.85E+18	2.38E+17
2.25	1.62E+18	1.42E+18	1.14E+18	6.60E+17	3.14E+17	1.29E+17
2.75	8.32E+17	7.66E+17	6.23E+17	3.19E+17	1.04E+17	1.25E+16
3.5	9.02E+15	8.35E+15	6.88E+15	3.69E+15	1.44E+15	4.50E+14
5	4.35E+11	2.33E+11	2.33E+11	2.32E+11	2.30E+11	2.26E+11
7	3.77E+10	3.76E+10	3.76E+10	3.74E+10	3.70E+10	3.65E+10
9.5	5.88E+09	5.87E+09	5.86E+09	5.83E+09	5.78E+09	5.69E+09
Total	8.92E+19	6.99E+19	5.16E+19	3.26E+19	2.18E+19	1.41E+19

Table 12.02-3-1—U.S. EPR Full-Core Bounding Photon Spectra at 4612 MW_t

(Page 3 of 3)

Photon Energy (MeV)	Photon Spectra (MeV/sec) as a Function of Shutdown Time (hrs)					
	5.0E+03 (0.57 yrs)	1.0E+04 (1.14 yrs)	2.0E+04 (2.28 yrs)	5.0E+04 (5.7 yrs)	1.0E+05 (11.4 yrs)	2.0E+05 (22.8 yrs)
0.01	1.49E+15	1.04E+15	7.36E+14	5.50E+14	4.53E+14	3.40E+14
0.025	2.68E+15	1.87E+15	1.36E+15	5.78E+14	1.47E+14	2.18E+13
0.0375	2.12E+16	1.41E+16	7.67E+15	3.41E+15	2.32E+15	1.48E+15
0.0575	1.15E+15	9.02E+14	7.14E+14	5.61E+14	5.11E+14	5.28E+14
0.085	9.96E+15	6.74E+15	3.55E+15	1.28E+15	5.69E+14	1.50E+14
0.125	6.99E+16	4.10E+16	1.92E+16	5.93E+15	3.24E+15	1.23E+15
0.225	5.31E+15	3.89E+15	3.19E+15	2.04E+15	1.12E+15	4.46E+14
0.375	2.21E+16	1.75E+16	1.25E+16	5.09E+15	1.35E+15	1.64E+14
0.575	2.80E+18	2.19E+18	1.53E+18	7.39E+17	4.37E+17	3.04E+17
0.85	3.66E+18	1.55E+18	9.00E+17	3.00E+17	6.11E+16	9.72E+15
1.25	3.52E+17	2.77E+17	1.88E+17	8.57E+16	4.01E+16	1.46E+16
1.75	3.21E+16	2.02E+16	9.85E+15	2.92E+15	1.53E+15	6.07E+14
2.25	7.32E+16	4.45E+16	1.67E+16	9.19E+14	9.48E+12	1.87E+11
2.75	1.42E+15	9.49E+14	4.32E+14	4.11E+13	1.00E+12	1.31E+11
3.5	2.67E+14	1.80E+14	8.25E+13	8.09E+12	3.83E+11	1.50E+11
5	2.17E+11	2.08E+11	1.97E+11	1.72E+11	1.39E+11	9.16E+10
7	3.51E+10	3.36E+10	3.17E+10	2.78E+10	2.25E+10	1.48E+10
9.5	5.47E+09	5.24E+09	4.95E+09	4.34E+09	3.51E+09	2.31E+09
Total	7.05E+18	4.17E+18	2.70E+18	1.15E+18	5.49E+17	3.34E+17

Part ii)

The U.S. EPR safety injection system (SIS) source strengths at various times following a loss of coolant accident (LOCA) are presented in Table 12.02-3-2. The tabulated spectra were based on the following:

a) A radionuclide mix at the time of the postulated LOCA containing the bounding core inventory for each individual radionuclide of importance for the following core parameters (as listed in U.S. EPR FSAR Tier 2, Table 15.0-14):

Power level:	4612 MW_t
U-235 fuel enrichment:	2 to 5 $^w/_o$
Fuel burn-up:	5 to 62 GWD/MTU
Core loading:	129 MTU (metric tons of uranium)

b) Core release fractions as defined in RG 1.183 for the alternative source term (AST) methodology.

c) Instantaneous transfer from the core of all released halogens and other particulates directly into the post-LOCA liquids in the in-containment refueling water storage tank (IRWST), which is the supply source for the SIS (i.e., combined gap and early in-vessel releases to the IRWST at t=0 hr, a conservative assumption).

d) Instantaneous evolution of all noble gases generated within the IRWST from halogen decay directly to the containment atmosphere.

e) 17 decay times spanning the range from 0 hrs to 1 year.

Table 12.02-3-2—U.S. EPR Post-LOCA Photon Spectra for Waterborne Sources

(Page 1 of 3)

Photon Energy (MeV)	Photon Spectra (MeV/s-m^3) as a Function of Post-LOCA Time (hrs)					
	0	0.1	0.2	0.5	1	2
0.01	9.75E+11	8.02E+11	7.22E+11	6.20E+11	5.58E+11	5.16E+11
0.025	1.23E+13	8.28E+12	7.13E+12	6.13E+12	5.46E+12	4.78E+12
0.038	6.91E+12	4.71E+12	3.99E+12	3.36E+12	2.96E+12	2.54E+12
0.058	1.87E+12	1.84E+12	1.83E+12	1.82E+12	1.81E+12	1.80E+12
0.085	8.61E+12	7.84E+12	7.20E+12	5.83E+12	4.53E+12	3.40E+12
0.125	3.38E+13	3.14E+13	2.98E+13	2.54E+13	1.85E+13	1.03E+13
0.225	3.12E+14	1.97E+14	1.77E+14	1.50E+14	1.25E+14	1.00E+14
0.375	8.16E+14	5.73E+14	5.48E+14	4.92E+14	4.31E+14	3.68E+14
0.575	3.72E+15	3.62E+15	3.54E+15	3.27E+15	2.85E+15	2.29E+15
0.85	6.89E+15	5.81E+15	5.38E+15	4.55E+15	3.56E+15	2.34E+15
1.25	8.05E+15	6.64E+15	6.41E+15	5.46E+15	4.02E+15	2.46E+15
1.75	2.03E+15	1.87E+15	1.85E+15	1.79E+15	1.63E+15	1.29E+15
2.25	1.54E+15	1.23E+15	1.15E+15	9.10E+14	5.78E+14	2.51E+14
2.75	1.08E+15	7.77E+14	7.22E+14	5.62E+14	3.57E+14	1.58E+14
3.5	1.10E+15	4.49E+14	2.54E+14	1.43E+14	8.75E+13	4.16E+13
5	1.24E+15	3.06E+14	7.57E+13	1.57E+13	1.39E+13	1.08E+13
7	3.18E+03	3.18E+03	3.18E+03	3.18E+03	3.18E+03	3.18E+03
9.5	4.95E+02	4.95E+02	4.95E+02	4.96E+02	4.96E+02	4.96E+02
Total	2.68E+16	2.15E+16	2.02E+16	1.74E+16	1.37E+16	9.33E+15

Table 12.02-3-2—U.S. EPR Post-LOCA Photon Spectra for Waterborne Sources

(Page 2 of 3)

Photon Energy (MeV)	Photon Spectra ($MeV/s\text{-}m^3$) as a Function of Post-LOCA Time (hrs)					
	5	10	20	24	50	100
0.01	4.72E+11	4.32E+11	3.85E+11	3.70E+11	2.94E+11	2.00E+11
0.025	4.21E+12	3.94E+12	3.62E+12	3.52E+12	2.95E+12	2.16E+12
0.038	2.07E+12	1.85E+12	1.71E+12	1.68E+12	1.51E+12	1.26E+12
0.058	1.76E+12	1.71E+12	1.60E+12	1.56E+12	1.33E+12	1.00E+12
0.085	2.75E+12	2.63E+12	2.49E+12	2.44E+12	2.17E+12	1.76E+12
0.125	5.21E+12	4.33E+12	3.75E+12	3.58E+12	2.68E+12	1.61E+12
0.225	7.71E+13	6.57E+13	5.53E+13	5.27E+13	4.24E+13	3.08E+13
0.375	3.15E+14	2.93E+14	2.71E+14	2.64E+14	2.36E+14	1.98E+14
0.575	1.64E+15	1.27E+15	9.82E+14	9.08E+14	6.22E+14	4.39E+14
0.85	1.11E+15	7.20E+14	5.53E+14	5.26E+14	4.39E+14	3.73E+14
1.25	1.33E+15	8.22E+14	4.05E+14	3.27E+14	1.60E+14	1.13E+14
1.75	7.26E+14	3.56E+14	1.36E+14	1.06E+14	7.40E+13	8.88E+13
2.25	8.26E+13	4.45E+13	1.87E+13	1.43E+13	5.85E+12	3.98E+12
2.75	4.28E+13	1.31E+13	3.26E+12	2.94E+12	4.03E+12	5.20E+12
3.5	1.32E+13	3.75E+12	3.45E+11	1.47E+11	4.16E+10	5.40E+10
5	5.03E+12	1.48E+12	1.29E+11	4.85E+10	8.49E+07	2.01E+04
7	3.18E+03	3.18E+03	3.18E+03	3.18E+03	3.17E+03	3.17E+03
9.5	4.96E+02	4.95E+02	4.95E+02	4.95E+02	4.95E+02	4.95E+02
Total	5.36E+15	3.61E+15	2.44E+15	2.21E+15	1.59E+15	1.26E+15

Table 12.02-3-2—U.S. EPR Post-LOCA Photon Spectra for Waterborne Sources

(Page 3 of 3)

Photon Energy (MeV)	Photon Spectra (MeV/s-m^3) as a Function of Post-LOCA Time (hrs)				
	200	500	720	4320	8766
0.01	1.09E+11	4.17E+10	2.74E+10	7.44E+09	6.86E+09
0.025	1.26E+12	4.12E+11	2.44E+11	5.07E+10	4.02E+10
0.038	9.61E+11	6.03E+11	5.02E+11	3.49E+11	3.31E+11
0.058	6.18E+11	2.30E+11	1.34E+11	2.86E+08	1.22E+08
0.085	1.21E+12	4.49E+11	2.27E+11	1.97E+09	1.28E+09
0.125	6.80E+11	1.71E+11	1.17E+11	1.69E+10	9.64E+09
0.225	1.79E+13	5.35E+12	2.71E+12	5.09E+10	4.29E+10
0.375	1.40E+14	5.02E+13	2.41E+13	3.77E+11	3.27E+11
0.575	3.57E+14	3.05E+14	2.93E+14	2.53E+14	2.22E+14
0.85	3.17E+14	2.58E+14	2.42E+14	1.92E+14	1.62E+14
1.25	8.29E+13	4.95E+13	3.84E+13	1.88E+13	1.58E+13
1.75	8.49E+13	4.50E+13	2.75E+13	8.02E+10	1.88E+10
2.25	2.35E+12	7.99E+11	4.82E+11	3.24E+10	1.70E+10
2.75	5.02E+12	2.67E+12	1.62E+12	2.03E+09	1.09E+09
3.5	5.26E+10	2.83E+10	1.74E+10	3.00E+08	2.09E+08
5	1.96E+04	1.95E+04	1.94E+04	1.85E+04	1.77E+04
7	3.17E+03	3.15E+03	3.14E+03	2.98E+03	2.86E+03
9.5	4.94E+02	4.91E+02	4.89E+02	4.65E+02	4.45E+02
Total	1.01E+15	7.18E+14	6.31E+14	4.65E+14	4.01E+14

Part iii)

The normal residual heat removal (RHR) system source strength is presented in Table 12.02-3-3 showing a series of decay times after shutdown. The tabulated spectra were based on the following:

a) The initial (un-decayed) RCS coolant inventory was assumed to correspond to the design-basis source term listed in U.S. EPR FSAR Tier 2, Table 11.1-2.

b) No dilution credit was taken into account for the extra coolant injected into the RCS to compensate for the coolant volume reduction induced by the cool-down.

c) The RCS coolant was assumed to be degasified prior to shutdown. Specifically, the noble-gas concentrations in the RHR piping were assumed to correspond to 1 percent of those during normal operation.

d) The coolant density was assumed to be 1 g/cc during the entire cool-down process by the RHR.

e) The time array for the post-shutdown decay was assumed to include the following time steps: 3, 6, 9 12, 15 and 18 hours. The typical time for RHR cool-down startup (based on a cool-down rate of 90°F per hour and coolant temperature reduction from 594°F at full power to 250°F) is about 4 hours.

Table 12.02-3-3—Photon Spectra for Residual Heat Removal System

Photon Energy (MeV)	Photon Spectra (MeV/sec-m^3) as a Function of Post-Shutdown Time (hrs) - Coolant at 1 (g/cc)					
	3	6	9	12	15	18
0.01	1.87E+07	1.84E+07	1.80E+07	1.77E+07	1.74E+07	1.72E+07
0.025	8.37E+07	7.89E+07	7.57E+07	7.32E+07	7.11E+07	6.93E+07
0.0375	5.76E+08	5.78E+08	5.79E+08	5.77E+08	5.74E+08	5.70E+08
0.0575	2.78E+07	2.73E+07	2.68E+07	2.64E+07	2.60E+07	2.56E+07
0.085	1.13E+09	1.13E+09	1.13E+09	1.13E+09	1.12E+09	1.12E+09
0.125	3.29E+08	3.59E+08	3.81E+08	3.92E+08	3.97E+08	3.96E+08
0.225	2.82E+09	3.23E+09	3.35E+09	3.29E+09	3.12E+09	2.90E+09
0.375	9.38E+09	9.04E+09	8.80E+09	8.60E+09	8.43E+09	8.28E+09
0.575	3.91E+10	3.31E+10	2.95E+10	2.69E+10	2.48E+10	2.30E+10
0.85	2.05E+10	1.55E+10	1.34E+10	1.23E+10	1.16E+10	1.10E+10
1.25	2.91E+10	2.19E+10	1.72E+10	1.38E+10	1.13E+10	9.44E+09
1.75	9.61E+09	6.79E+09	4.92E+09	3.60E+09	2.64E+09	1.94E+09
2.25	1.85E+09	1.18E+09	8.19E+08	5.88E+08	4.32E+08	3.25E+08
2.75	3.34E+09	2.86E+09	2.48E+09	2.16E+09	1.88E+09	1.64E+09
3.5	1.73E+07	4.66E+06	3.09E+06	2.33E+06	1.85E+06	1.53E+06
5	2.12E+06	7.92E+05	3.93E+05	2.01E+05	1.08E+05	6.12E+04
7	9.65E-04	9.65E-04	9.65E-04	9.64E-04	9.64E-04	9.64E-04
9.5	1.51E-04	1.51E-04	1.51E-04	1.51E-04	1.51E-04	1.51E-04
Total	1.18E+11	9.57E+10	8.26E+10	7.34E+10	6.63E+10	6.08E+10

FSAR Impact:

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

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.

Response to Question 12.03-12.04-3:

1. Although ANSI/ANS 6.8.1 was withdrawn in May 1992, U.S. EPR calibration techniques will be based on the guidance provided in ANSI/ANS 6.8.1 and will be performed in a manner consistent with as low as reasonably achievable (ALARA) principles. The calibration accuracy will be defined in the calibration procedure used. Accuracy over the range of measurement will be documented by the manufacturer for each type of monitor. Additionally, for fixed area and airborne monitors used for accident monitoring instrumentation, the U.S. EPR will use the calibration guidance of IEEE 497-2002. U.S. EPR FSAR Tier 2, Table 7.1-2 denotes the applicability of this code to the radiation monitoring system. Additional detail on these monitors is vendor-dependent.

FSAR Impact:

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

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:

Overview of Aeroball Measurement System:

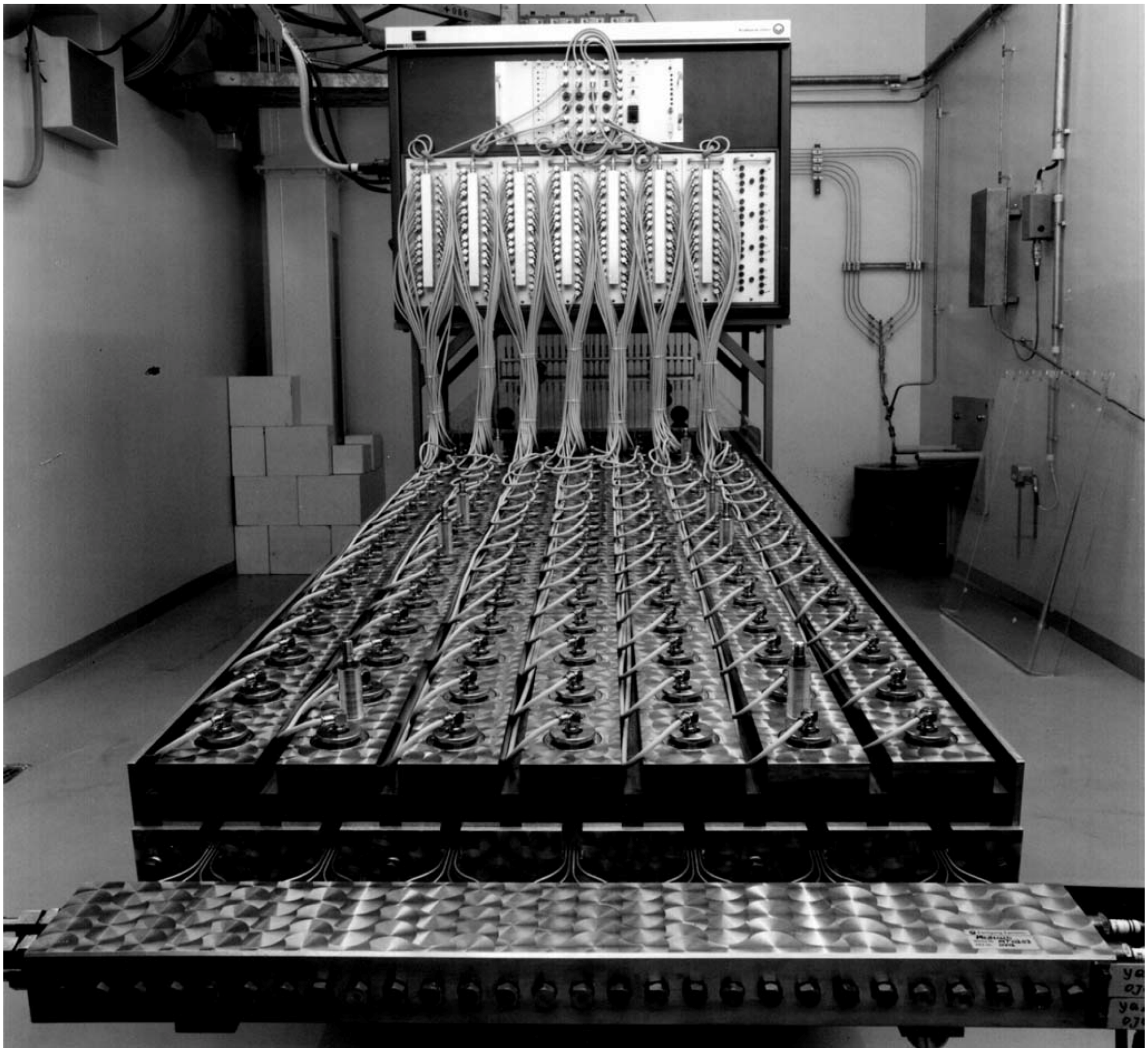
The Aeroball Measurement System (AMS) is an electromechanical computer-controlled, fully automated, online flux mapping measurement system based on movable activation probes. The AMS has lances that extend downward into the core, uses a transport system located outside of the reactor vessel, and employs measuring equipment located in a remote area. The movable activation probes, or “aeroballs”, are steel balls, composed of carbon, chromium, iron and vanadium. The useful neutron sensitive material is the vanadium (specifically V-51) which undergoes the following nuclear reaction:

$$V^{51} + n_0^1 \rightarrow V^{52} + \beta + \gamma \rightarrow Cr^{52}$$

The vanadium used in the balls produces a gamma decay signature readily discernable by the measurement software. When exposed to a neutron flux, the V-51 used in the balls absorbs a neutron and reaches a higher energy state to become isotope V-52. With a half life of 3.76 minutes, V-52 undergoes β- decay to Cr-52. Gamma radiation that is given off during this decay is measured by the AMS and the energy of the emitted γ used in the activity determination process is approximately 1.43 MeV.

The measured activity distribution along the Aeroball measurement table stacks (see Figure 12.03-12.04-4-1) is proportional to the neutron flux density and thus, to the power density in the measured region of the reactor core. The result is a three dimensional power density distribution for the reactor core which can be used to calibrate the self-powered neutron detectors. The balls are designed to be used for the life of the plant without the need for replacement since there is negligible depletion of V-51 during each measurement. Nevertheless, the balls can be replaced if Aeroball transport problems occur.

Figure 12.03-12.04-4-1—Example of AMS Measurement Table Currently in Use for KONVOI-Type Plants



Part i)

The AMS will only be used during power operation to measure neutron flux. During a refueling outage, the AMS lances remain covered by water in the same manner that the core and fuel elements remain covered by water. Should personnel need to be in the proximity of the ball stops, tubes containing the movable balls, or any other components of the AMS within containment during a refueling outage, they would not receive radiation doses in excess of occupational dose limits specified in 10 CFR Part 20 because the main isotopic constituents of V-52 activity decays away after 45 minutes and after 24 hours for Mn-56 (personnel are not permitted into these areas of containment during these timeframes after power operation). Furthermore, should the balls become incapable of movement, the irradiated ball isotopes will decay rapidly and will not pose a radiological hazard to personnel attempting to repair the system.

Should there be a need to replace the Aeroballs, this would be accomplished during a plant outage. The AMS measurement table has a manifold connection that is used whenever new Aeroballs are installed into the system or when changing out the Aeroballs. Radiation from the removed Aeroballs will not be a concern to personnel 24 hours after shutdown because of the short half-lives of the activation products associated with the materials of an Aeroball.

Historically, Aeroball systems have been operated for approximately 30 years (i.e., over 220 fuel cycles) at over 12 plants with typically 28 Aeroball probes in each cycle. Table 12.03-12.04-4-1 contains a listing of nuclear power plants that have used the AMS. Table 12.03-12.04-4-2 contains a list of abnormal occurrences involving the AMS.

Table 12.03-12.04-4-1—Nuclear Power Plants Using the Aeroball Measurement System

	Plant Name	Plant Acronym	Capacity (MW)	First Power Generation	End of Power Generation
	Germany				
1	Biblis A	KWB A	1,225	8/25/1974	Still In Operation
2	Biblis B	KWB B	1,300	4/25/1976	Still In Operation
3	Brikdorf	KBR	1,440	10/14/1986	Still In Operation
4	Emsland	KKE	1,400	4/19/1988	Still In Operation
5	Grafenrheinfeld	KKG	1,345	12/18/1981	Still In Operation
6	Grohnde	KWG	1,430	9/4/1984	Still In Operation
7	Isar2	KKI 2	1,475	1/22/1988	Still In Operation
8	Neckarwestheim1	GKN 1	840	6/3/1976	Still In Operation
9	Neckarwestheim2	GKN 2	1,365	1/3/1989	Still In Operation
10	Obrigheim	KWO	357	10/29/1968	5/11/2005
11	Philppsburg 2	KKP 2	1,458	12/17/1984	Still In Operation
12	Unterweser	KKU	1,410	10/1/1978	Still In Operation
	Netherlands				
13	Borssele	KCB	480	7/4/1973	Still In Operation
	Spain				
14	Trillo 1	CNT	1,066	5/23/1988	Still In Operation
	Switzerland				
15	Gosegen	KKG	1,020	2/2/1979	Still In Operation
	Brasil				
16	Angra 2		1,350	7/21/2000	Still In Operation

Table 12.03-12.04-4-2—Abnormal Operational Occurrences Associated with AMS Probes

Plant:	Biblis A	KKP 2	KKP 2	Other Reactor Plants
Year:	1975/76	1989/90	1990/91	1975 to Present
Time Since Beginning of Cycle:	Cycle 1 245 EFPD	Cycle 5 70 EFPD	Cycle 6 6 EFPD	See Note 1
Number of Failed Probes:	14 of 32	21 of 28	21 of 28	
Reason:	Wrong lubrication	Excessive residual moisture after N_2 tank hydro test	Subsequent failure of same probes in next cycle	

NOTE:

1. No abnormal operation of AMS probes has been observed for any reactor plants other than those addressed in this table.

Part ii)

Aeroballs are spheres of 1.7 mm diameter with approximately 2500 balls per stack. When at the rest position, the ball stacks will be above the solenoid ball stops which are located above the reactor pressure vessel at the level of the cable bridge. When the ball stops (which are located inside containment) are opened, the ball stacks move pneumatically by means of nitrogen gas to their activation positions in the reactor core. Figure 12.03-12.04-4-2 illustrates the AMS pneumatic system. There are 40 columns of ball stacks, divided into 4 subsystems, with a length approximately equal to the active core height. The activity distribution along the stacks will be proportional to the neutron flux density and thus to the power density at the place of activation. After a defined irradiation time, the ball stacks leave the core via nitrogen gas through ball stops and pass into the measuring table. In the measuring compartment, the ball transport tubes are connected to a measuring table comprised of several detector beams (see Figure 12.03-12.04-4-1). All beams are arranged on the measuring table with 4 ball transport tubes grouped together in parallel under each detector beam to form 4 subsystems. The ball stacks of one subsystem are measured simultaneously with the measurements being performed individually for each stack. During the measurements, the activated ball stacks of the other three subsystems are in the rest position which is located above the reactor pressure vessel. Figure 12.03-12.04-4-3 shows the location of the AMS relative to the reactor pressure vessel.

An illustration of a solenoid ball stop is shown in Figure 12.03-12.04-4-4. There are 40 solenoid ball stops that are situated in the ball transport tubes between the lance and the measuring table with all ball stops located close to the lance. The ball stop guide tube is closed with a piston mounted in a pressure-tight housing, but remains permeable to the nitrogen gas. The ball stop forms a gateway which is opened or closed for the ball stack movement. When it is open, the balls are permitted to be transported in either direction (i.e., to the core or to the measuring table). When the stop is closed, a defined wait or rest position located above the reactor pressure vessel is used for the ball stack. At the wait position, the ball stack is located below the ball stop (pressure applied in the direction measuring table) whereas at the rest position, the ball stack is located above the ball stop (pressure applied in the direction lance or no pressure applied), see Figure 12.03-12.04-4-2.

Shielding is not required at the wait or rest position to protect personnel due to the short half-lives of the activated materials in the Aeroballs, the limited access to this portion of containment above the reactor pressure vessel during operations, and the minimum wait time required prior to accessing this portion of containment after shutdown. In addition, access to the room where the measurements are taken is limited during counting periods.

Figure 12.03-12.04-4-2—AMS Pneumatic System

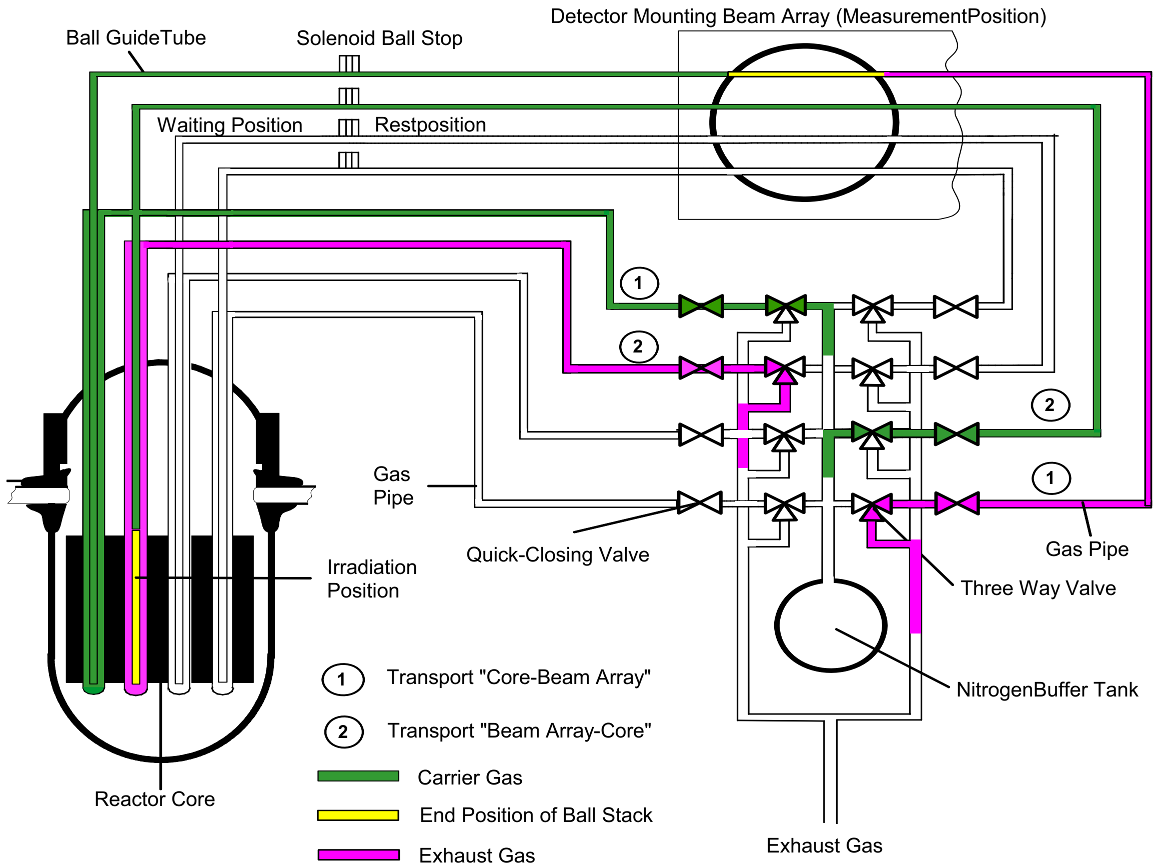


Figure 12.03-12.04-4-3—Location of AMS Instrumentation

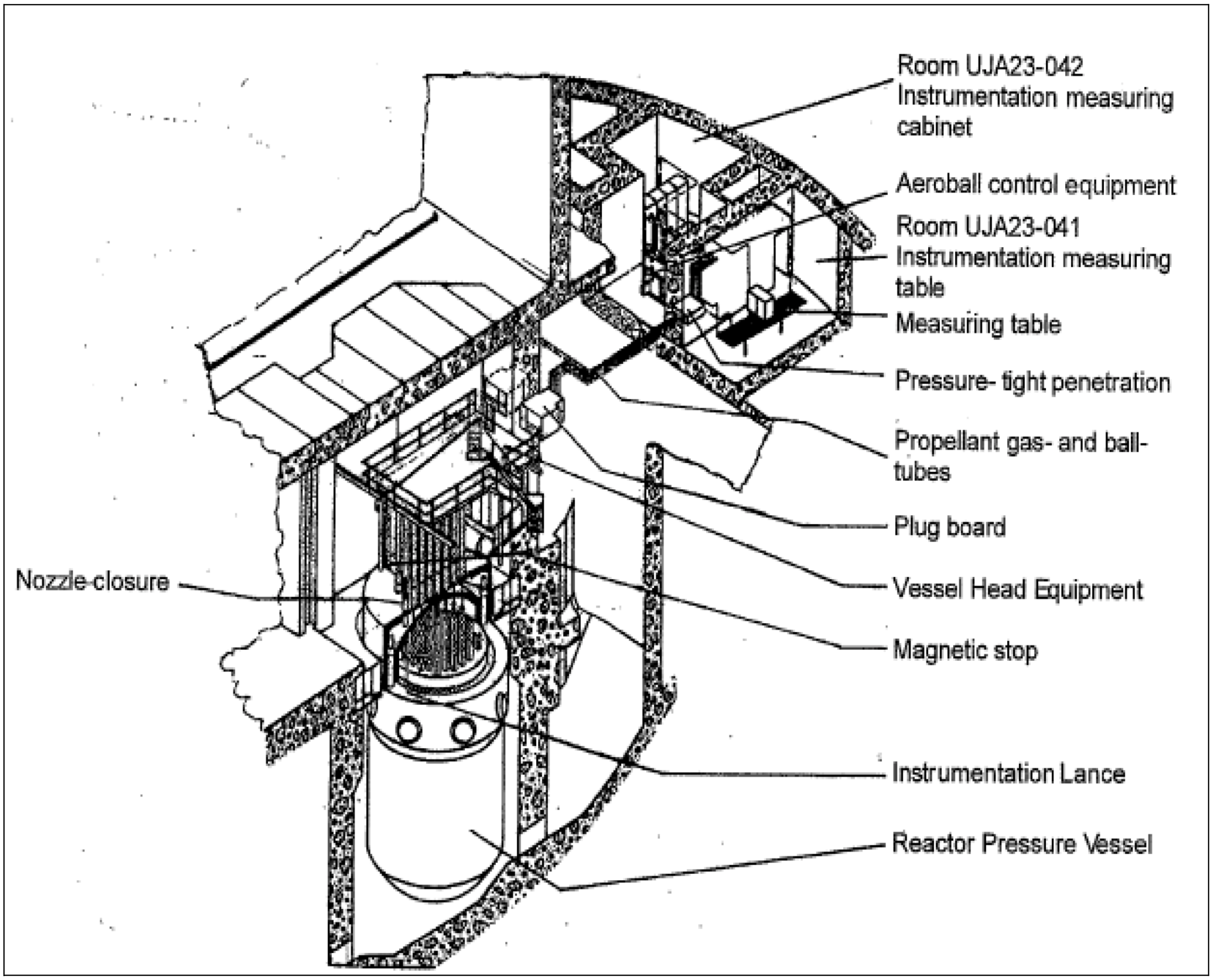
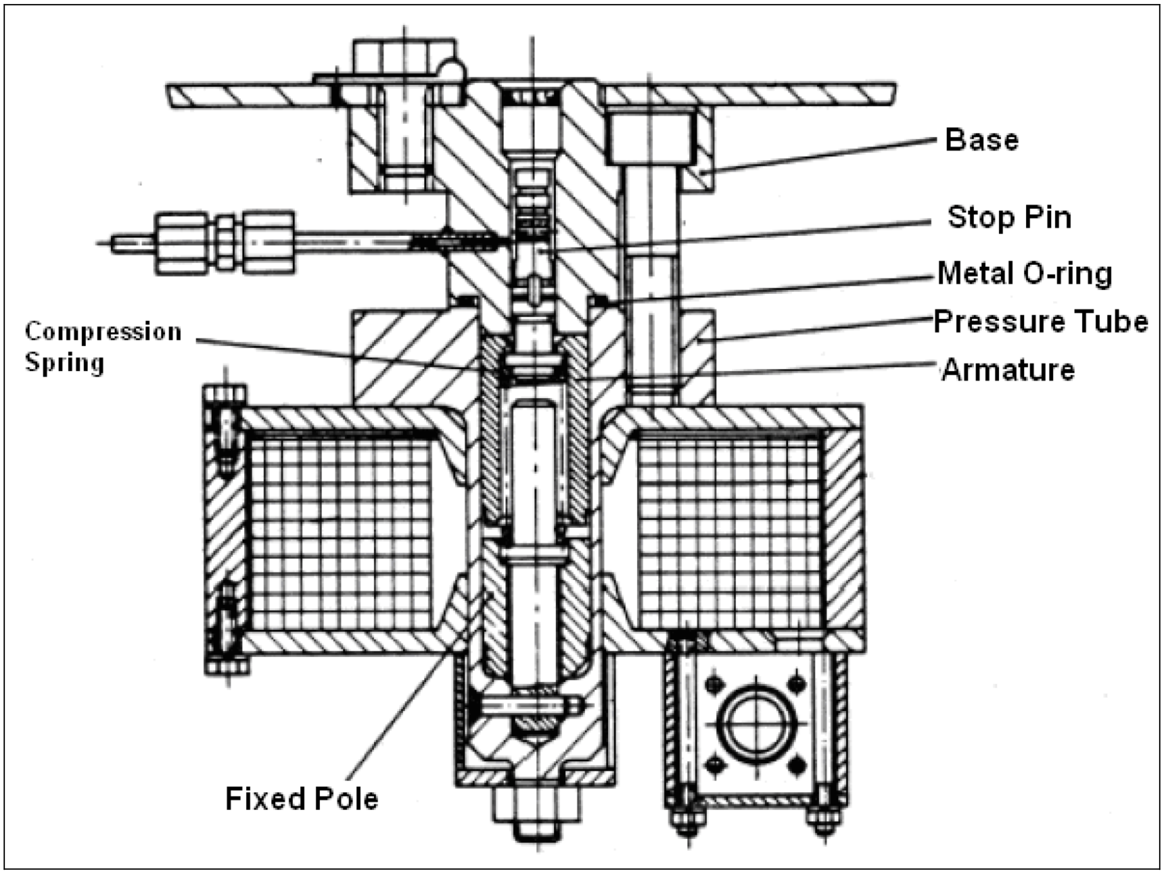


Figure 12.03-12.04-4-4—Solenoid Ball Stop



Part iii)

Each Aeroball is 1.7 mm in diameter and the typical composition of each Aeroball is:

weight % Cr-50 in Aeroball: 14.5

weight % Fe-56 in Aeroball: 83.36

weight % V-51 in Aeroball: 1.54

weight % C-12 in Aeroball: 0.6

The neutron capture reactions include:

$$ {}^{51}_{23}V \xrightarrow{(n,\gamma)} {}^{52}_{23}V \xrightarrow[3.76\,\mathrm{min}]{\beta,\gamma} {}^{52}_{24}Cr \qquad (\gamma = 1434.1\,keV) $$

$$ {}^{50}_{24}Cr \xrightarrow{(n,\gamma)} {}^{51}_{24}Cr \xrightarrow[27.7\,day]{\varepsilon,\gamma} {}^{51}_{23}V \qquad (\gamma = 320.1\,keV) $$

$$ {}^{56}_{26}Fe \xrightarrow{n} {}^{57}_{26}Fe $$

$$ {}^{12}_{6}C \xrightarrow{n} {}^{13}_{6}C $$

$$ {}^{55}_{25}Mn \xrightarrow{(n.\gamma)} {}^{56}_{25}Mn \xrightarrow[2.578\,hr]{\beta,\gamma} {}^{56}_{26}Fe \quad (\gamma = 846.8,\ 1810.8,\ 2113.1\,keV) $$

The V-52 activity decays away after 45 minutes and Mn-56 after 24 hours.

The content of the alloy elements with their tolerances are specified in detail and verified after fabrication by chemical analysis. During commissioning, two ball stacks will be activated so that they reach saturation. Their activity will then be measured to establish time dependence and decay constants of the material components.

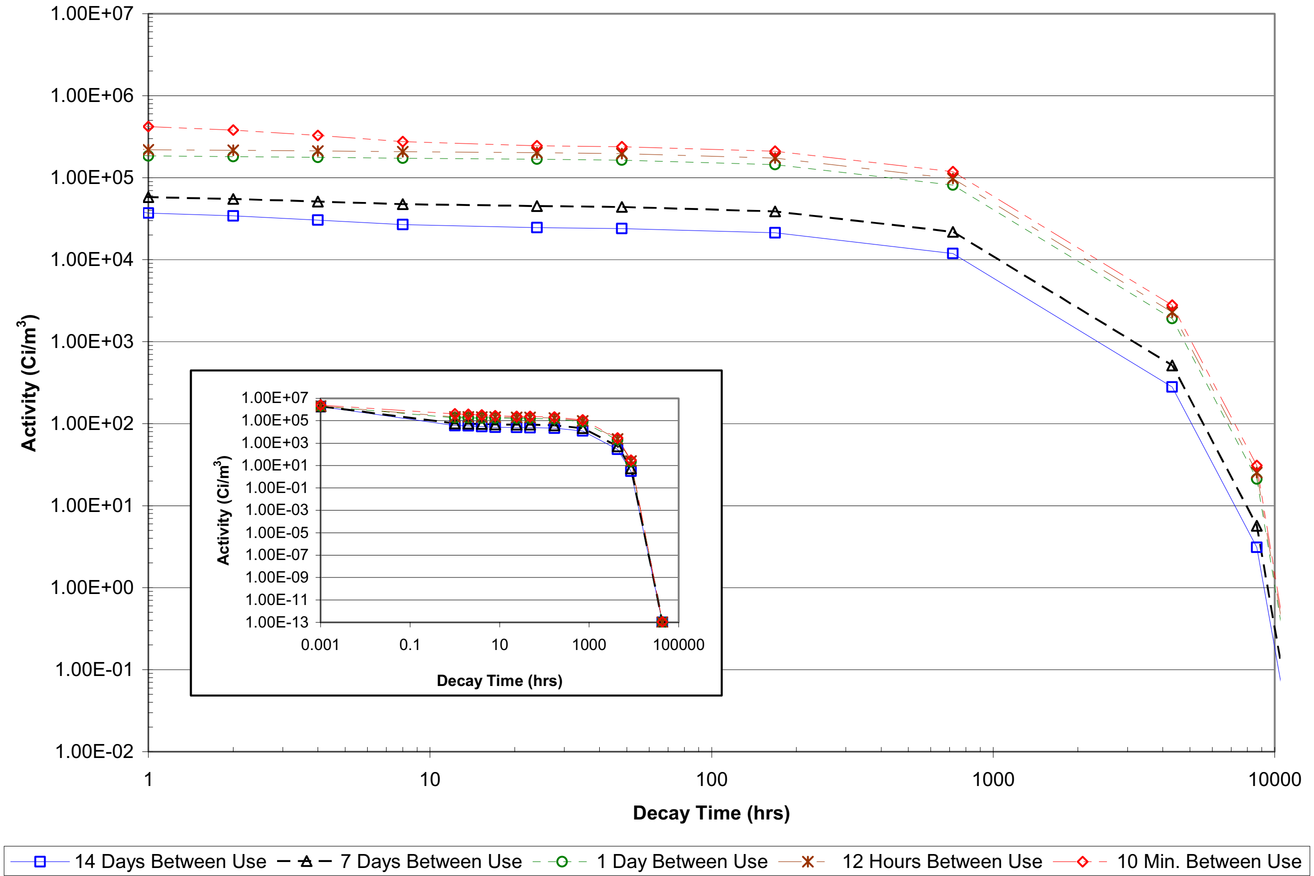
The bounding total activity for the Aeroballs was established by considering 34 consecutive irradiation cycles with each cycle consisting of a 4 minute irradiation followed by a 10 minute decay period after the irradiation. The conservative action of applying this minimal 10 minute decay time between irradiations 34 consecutive times will bound all conceivable future use of this system. Note that the typical decay period between irradiations is expected to be 14 days. Table 12.03-12.04-4-3 presents the total activity of the Aeroballs as a function of decay time after the application of these 34 irradiation cycles. Figure 12.03-12.04-4-5 shows the impact on the total activity as a function of decay time (or time between usage) after multiple irradiation cycles with 4 minute irradiations.

Table 12.03-12.04-4-3—Photon Release Rate for Aeroballs

Energy Group (MeV)	Photon Release Rate (photon/m^3-sec)								
	Discharge	**1 Hour**	**2 Hours**	**4 Hours**	**8 Hours**	**1 Day**	**2 Days**	**1 Week**	**1 Month**
1.00E-02	5.56E+15	1.32E+15	1.25E+15	1.16E+15	1.07E+15	1.02E+15	9.90E+14	8.74E+14	4.91E+14
2.50E-02	1.00E+15	6.07E+13	4.64E+13	2.71E+13	9.25E+12	1.25E+11	2.08E+08	1.47E+06	2.37E+02
3.75E-02	6.70E+14	4.07E+13	3.11E+13	1.82E+13	6.20E+12	8.40E+10	1.39E+08	9.21E+05	1.48E+02
5.75E-02	1.01E+15	6.06E+13	4.63E+13	2.71E+13	9.23E+12	1.25E+11	2.06E+08	1.22E+06	1.97E+02
8.50E-02	6.43E+14	3.86E+13	2.95E+13	1.72E+13	5.87E+12	7.96E+10	1.30E+08	6.46E+05	1.04E+02
1.25E-01	4.42E+14	2.64E+13	2.02E+13	1.18E+13	4.02E+12	5.45E+10	8.86E+07	3.62E+05	5.83E+01
2.25E-01	6.57E+14	3.88E+13	2.97E+13	1.73E+13	5.91E+12	8.05E+10	4.07E+08	4.19E+07	6.80E+03
3.75E-01	1.11E+15	7.96E+14	7.90E+14	7.82E+14	7.73E+14	7.57E+14	7.39E+14	6.52E+14	3.67E+14
5.75E-01	2.41E+14	1.20E+13	9.15E+12	5.34E+12	1.82E+12	2.47E+10	3.91E+07	2.70E+03	4.34E-01
8.50E-01	8.29E+15	6.21E+15	4.75E+15	2.77E+15	9.46E+14	1.28E+13	2.58E+10	8.28E+08	1.33E+05
1.25E+00	8.47E+16	1.20E+13	8.16E+12	4.77E+12	1.64E+12	3.51E+10	8.99E+09	1.34E+09	2.16E+05
1.75E+00	2.38E+15	1.77E+15	1.35E+15	7.90E+14	2.70E+14	3.66E+12	5.78E+09	5.66E-05	4.09E-13
2.25E+00	1.11E+15	8.50E+14	6.50E+14	3.80E+14	1.30E+14	1.76E+12	2.78E+09	2.72E-05	0.00E+00
2.75E+00	1.56E+14	1.19E+14	9.10E+13	5.32E+13	1.81E+13	2.46E+11	3.89E+08	3.81E-06	0.00E+00
3.50E+00	1.34E+13	1.02E+13	7.80E+12	4.55E+12	1.55E+12	2.11E+10	3.33E+07	3.26E-07	0.00E+00
5.00E+00	1.55E-03	8.21E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7.00E+00	7.98E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
9.50E+00	5.05E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total (photon/m^3-sec)	1.08E+17	1.14E+16	9.11E+15	6.07E+15	3.26E+15	1.79E+15	1.73E+15	1.53E+15	8.58E+14
Total (MeV/m^3-sec)	1.21E+17	1.10E+16	8.48E+15	5.08E+15	1.93E+15	3.16E+14	2.87E+14	2.53E+14	1.42E+14

Figure 12.03-12.04-4-5—Total Activity for Aeroballs as Function of Decay Time for Various Times Between Usage

(inset provides full range)



Part iv)

The carrier gas is high purity, naturally occurring nitrogen which is comprised of N-14 and N-15. High purity nitrogen is used to ensure that the probability of trace contaminants that could become activated, such as argon, is greatly reduced. In addition, the half-lives of any activation products of this nitrogen are very short (on the order of 10 seconds) and hence, will rapidly decay. The U.S. Boiling Water Reactor experience with the use of nitrogen as a carrier gas in the active core region shows that it will produce an insignificant amount of radioactive gases.

Any nitrogen gas remaining in the AMS tubes following a measurement will occupy approximately 0.02 cubic ft in the active region of the core. Additionally, at least every 15 EFPD, an AMS reading is performed such that nitrogen gas in the tubes is swept out of the active region of the core. Thus, an insignificant amount of high purity, naturally occurring nitrogen becomes activated to N-16. Operational experience with Konvoi plants has shown that there has been insignificant dose to personnel as a result of the Aeroball measurements.

The valve rack will be situated in the AMS equipment room adjacent to the AMS measuring table in the measuring room. It will comprise all valves necessary for ball transport control. The nitrogen gas pipes leading to the measuring table will be routed through this valve rack and placed parallel to the ball transport tubes in the neighboring AMS measuring table compartment. The nitrogen used to move the balls will be directed to the four pneumatic subsystems via pressure reducers, buffer tanks with pressure indicators and switches, solenoid valves, filters, and branches. Each of the four subsystems will be equipped with a separate solenoid valve control system. Gas supply piping leading to the individual subsystems will branch into two three-way valves. One of these two valves will control the nitrogen gas train leading to the instrumentation lance, whereas the other valve will control the nitrogen gas train leading to the measuring table. When the Aeroballs enter the measuring table, the nitrogen is filtered and exhausted into the AMS equipment room where the valve rack is installed. Whenever personnel are in the AMS equipment area, the HVAC for that area will be aligned to a filtered exhaust alignment.

No containment entries are necessary during the cycle until the last 2 weeks during which outage equipment staging is performed. Therefore, it is anticipated that personnel will not be in the AMS equipment rooms during operation. Additionally, the doors in the AMS equipment area and the measuring room are both locked, thereby providing additional restricted entry for personnel. When a measurement is performed, an automatic alarm sounds one minute before the Aeroball stacks enter the measuring room and the AMS measuring room is automatically locked, thereby preventing personnel entry. A crash bar permits personnel inside of the measuring room unimpeded egress from the measurement room.

FSAR Impact:

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

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:

The minimum wall thickness required to maintain dose rates in occupied areas adjacent to the fuel transfer tube less than 100 rads per hour is 2.33 ft. Figures 12.03-12.04-6-1 through 12.03-12.04-6-4 show this value in a box on the credited walls and floors. The estimated dose rates in the spent fuel transfer tube compartment and occupied adjacent compartments are shown in these figures, represented by an 'X' where the contact dose rate is estimated along with a value representing the estimated dose rate in rads per hour. These dose rates are also presented in Table 12.3-12.04-6-1.

Note that Figures 12.03-12.04-6-1 through 12.03-12.04-6-4 are magnified portions of the following structural drawings in U.S. EPR FSAR Tier 2, Chapter 3, Appendix 3B:

- Figure 12.03-12.04-6-1 is taken from Figure 3B-6.
- Figure 12.03-12.04-6-2 is taken from Figure 3B-14.
- Figure 12.03-12.04-6-3 is taken from Figure 3B-19.
- Figure 12.03-12.04-6-4 is taken from Figure 3B-28.

Design features control access to very high radiation areas through which the fuel assembly passes during spent fuel transfer. To control access during fuel transfers, access doors are used to limit entry to compartment UJA15-020 (Room 20 in Figure 12.03-12.04-6-1) from UJA15-013, UFA16-023 (Room 38 in Figure 12.03-12.04-6-3) from UFA15-096 (Room 37 in Figure 12.03-12.04-6-3), and the annulus entries. The design features include double locks, local and remote alarms, and video surveillance in compliance with 10 CFR 20.1602. These features also include crash-bars to permit the unimpeded exit of personnel in the affected area.

Table 12.03-12.04-6-1—Compartment Dose Rates Adjacent to the Fuel Transfer Tube

Compartment	Description	Direction	Shielding	Contact Dose Rate (rads/hr)
UJA15-021	**Fuel Transfer Tube**	**source**	none	1E+05 at 1 m
UJA15-017	Core Internals Storage	north	concrete	inaccessible
UJA15-016	Loop 4 Annular Area	east	water, concrete	2E-04
UJA15-016	Loop 4 Annular Area	east	concrete	2E-02
UJB15-003	Annulus 144º - 216º	east	concrete	2E-02
UFA16-023	Transfer Pit	south	none	fuel transfer tube
UJB15-003	Annulus 144º - 216º	west	concrete	2E-01
UJA15-020	Access to Transfer Tube	west	concrete labyrinth	2E-02
UJA15-020	Access to Transfer Tube	west	water without labyrinth	1E+03
UJA15-013	Loop 1 Annular Area	west	concrete	2E-05
UJA15-025	Fuel Pool Skimming Valve Room	west	water, concrete	8E-05
UJA15-023	Instrument Lance Storage	west	concrete	inaccessible
UJA18-013 UJA18-016	Loop 1&4 Annular Area	ceiling	concrete	6E-02
UJA11-016	Loop 1&4 Annular Area	floor	concrete	5E-03
UJA07-017	Venting Area for Spreading Compartment	floor	concrete	inaccessible
UFA16-023	**Transfer Pit**	**source**	none	1E+05 at 1 m
UJA15-021	Fuel Transfer Tube	north	none	fuel transfer tube
UFA15-096	Access to Transfer Pit	east	concrete	2E-02
UFA13-084	Volume Control Tank	east	concrete	2E-02
UFA13-083	Access to Volume Control Tank	east	concrete	2E-02
UFA15-022	Spent Fuel Pool	south/west	concrete	inaccessible
UFA15-045	Piping Penetration Area	west	concrete	2E-01
UFA21-045	Piping Penetration Area	ceiling	water, concrete	1E-06
UFA10-095	Piping Penetration Area	floor	water, concrete	3E-03
UFA10-085	CVCS Pump Room	floor	concrete	2E-04

Figure 12.03-12.04-6-1—Reactor Building Fuel Transfer Map +5.15 m Elevation

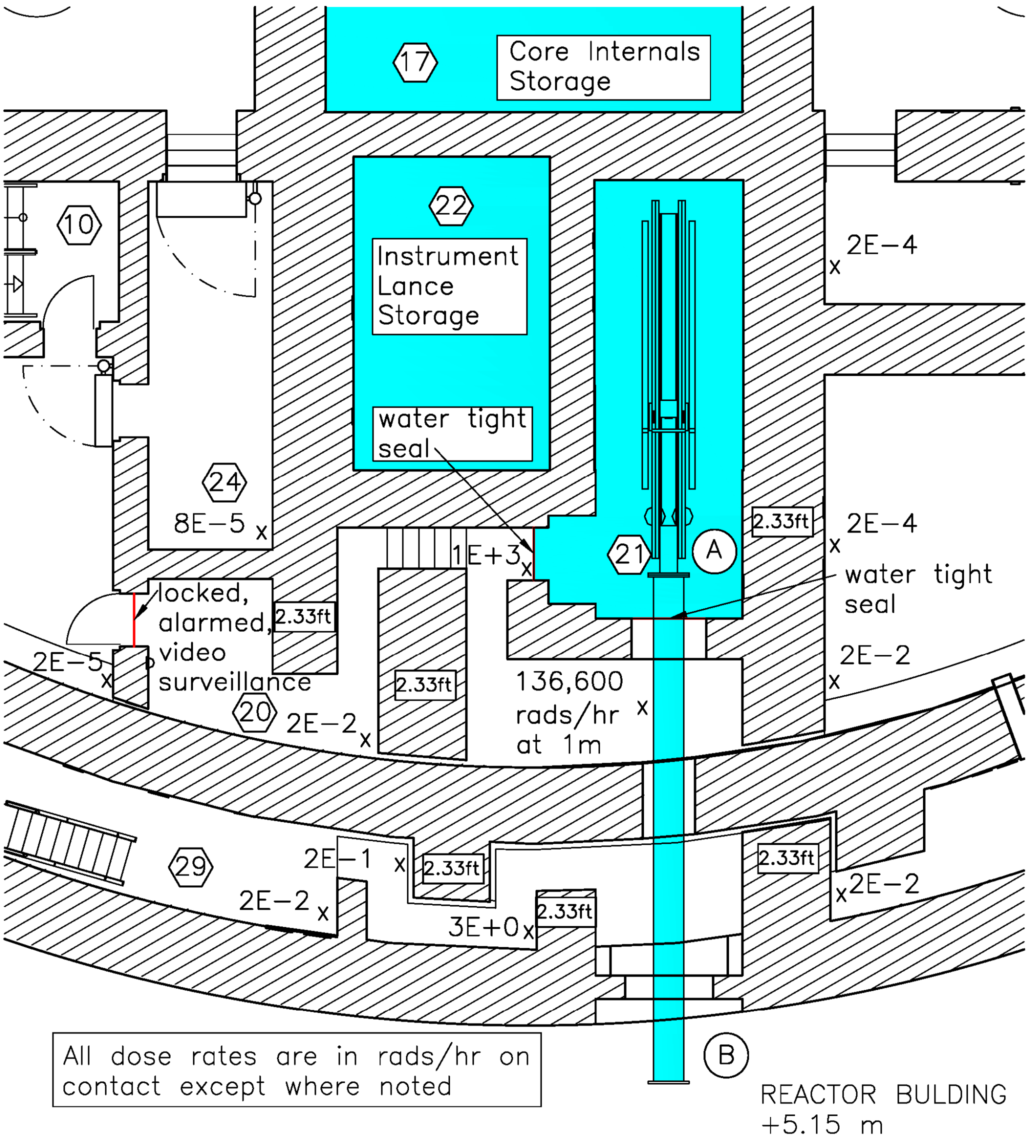
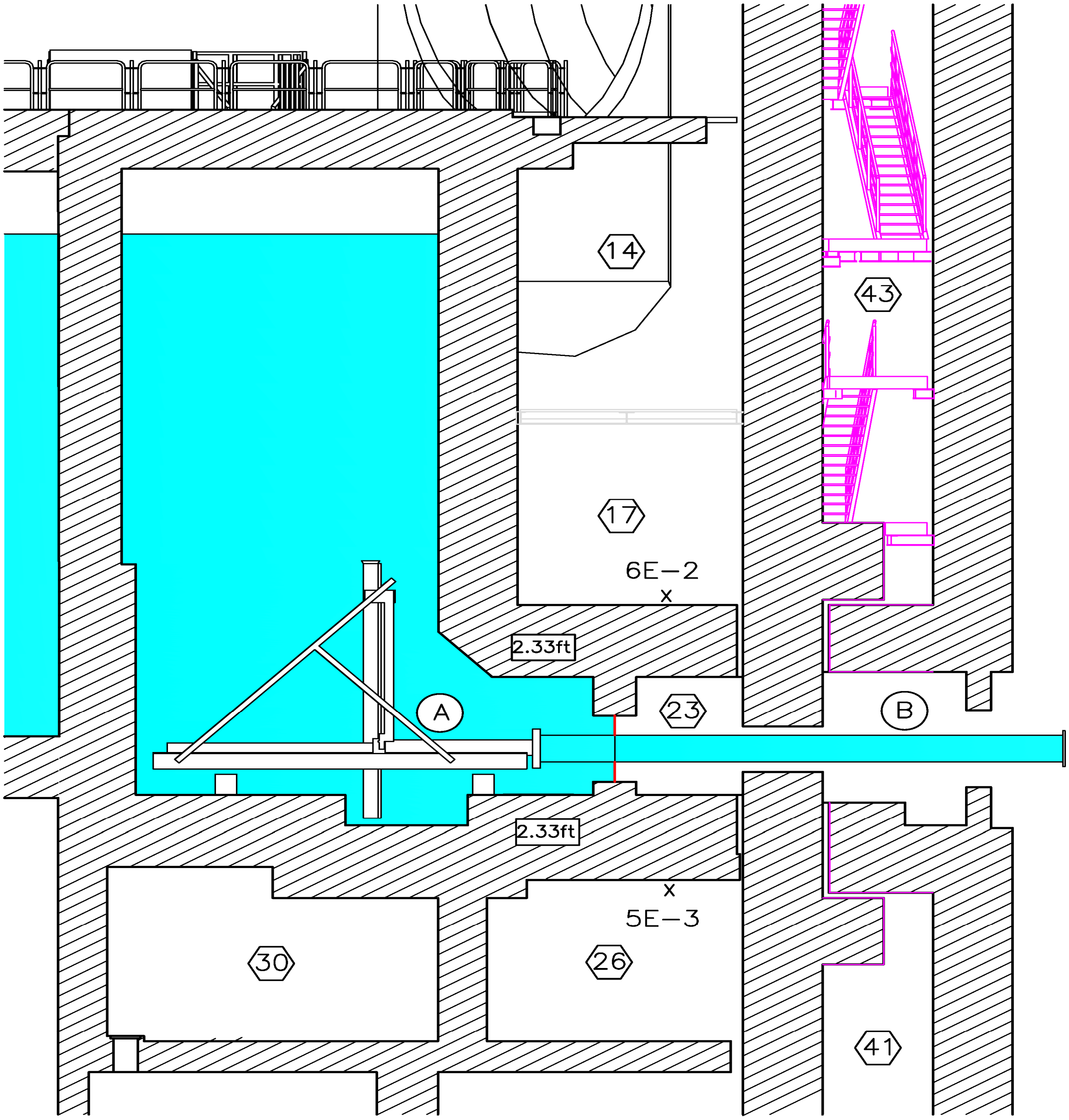


Figure 12.03-12.04-6-2—Reactor Building Fuel Transfer Map Section C-C



All dose rates are in rads/hr on contact except where noted

REACTOR BULDING
SECTION C−C

Figure 12.03-12.04-6-3—Fuel Building Fuel Transfer Map +3.70 m Elevation

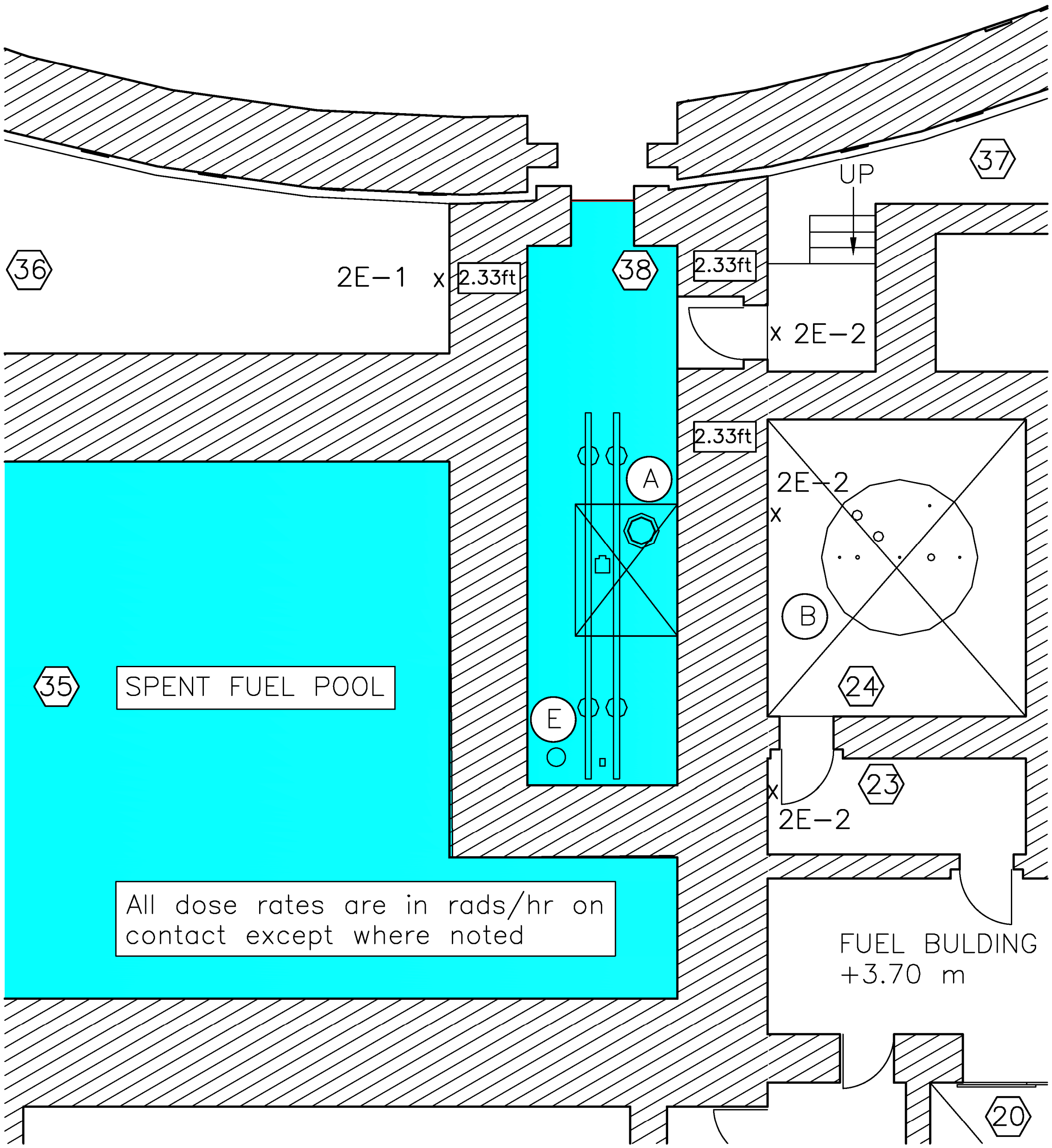
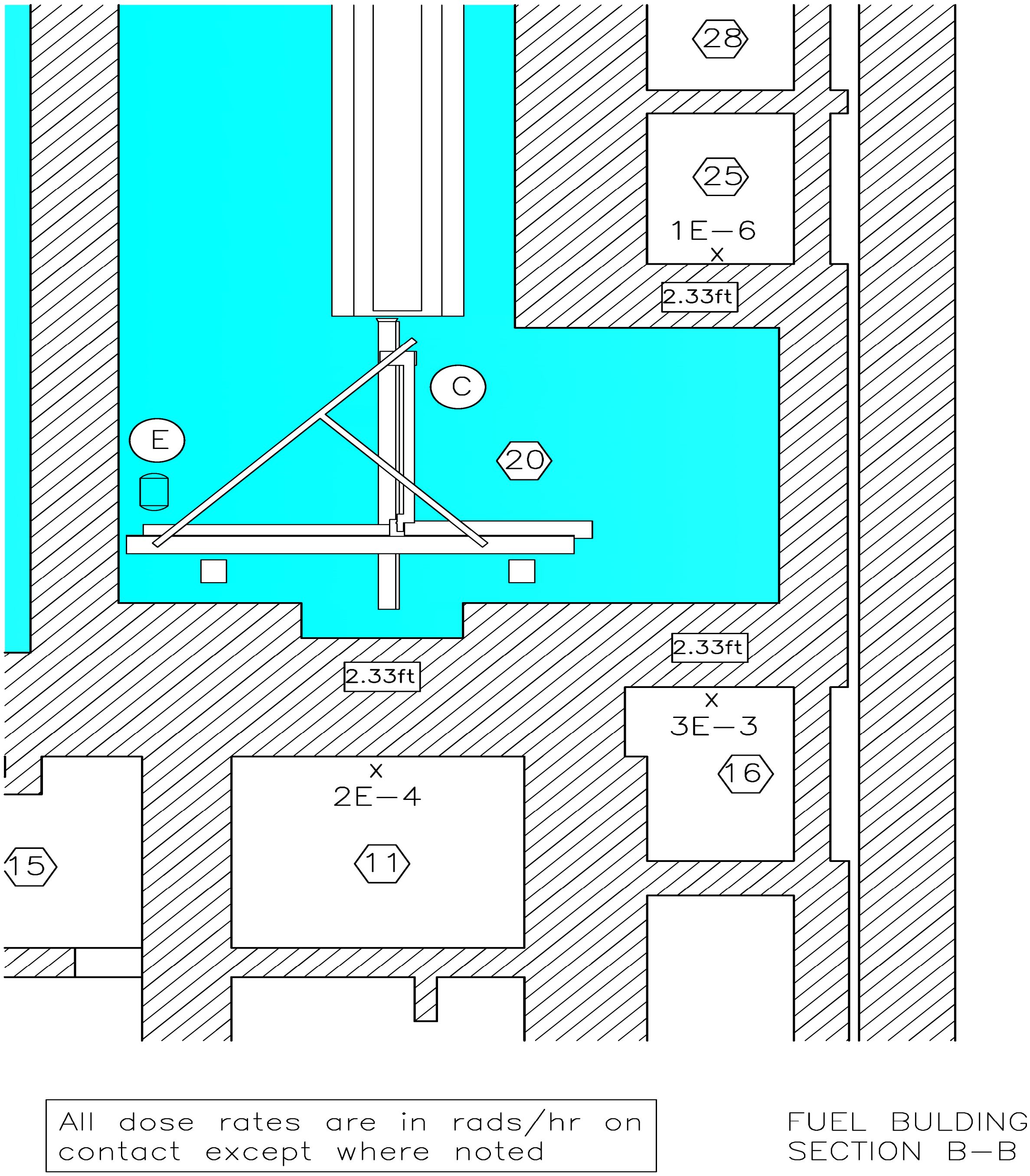


Figure 12.03-12.04-6-4—Fuel Building Fuel Transfer Map Section B-B



FSAR Impact:

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