CHAPTER 11

RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

This section addresses the sources of radioactivity that are treated by the liquid and gaseous radwaste systems. Radioactive materials are generated within the core (fission products) and have the potential of leaking to the reactor coolant system by way of defects in the fuel cladding. The core radiation field also results in activation of the coolant to form N-16 from oxygen and the activation of corrosion products in the reactor coolant system.

Two source terms are presented for the primary and the secondary coolant. The first is a conservative, or design basis, source term that assumes the design basis fuel defect level. This source term serves as a basis for system design and shielding requirements.

The second source term is a realistic model. This source term represents the expected average concentrations of radionuclides in the primary and the secondary coolant. These values are determined using the model in the PWR-GALE code (Reference 1) and which provides the bases for estimating typical concentrations of the principal radionuclides that are expected to occur. This source term model reflects the industry experience at a large number of operating PWR plants.

11.1.1 Design Basis Reactor Coolant Activity

11.1.1.1 Fission Products

For the design basis source term it is assumed that there is a significant fuel defect level, well above that anticipated during normal operation. It is assumed that small cladding defects are present in fuel rods producing 0.25 percent of the core power output (also stated as 0.25 percent fuel defects). The defects are assumed to be uniformly distributed throughout the core.

The parameters used in the calculation of the reactor coolant fission product concentrations, including pertinent information concerning the fission product escape rate coefficients, coolant cleanup rate, and demineralizer effectiveness, are listed in Table 11.1-1. Since the fuel defects are assumed to be uniformly distributed in the core, the fission product escape rate coefficients are based on average fuel temperature.

The determination of reactor coolant activity is based on time-dependent fission product core inventories that are calculated by the ORIGEN code (Reference 2).

The fission product activity in the reactor coolant is calculated using the following differential equations.

For parent nuclides in the coolant:

$$\frac{dN_{c_p}}{dt} = \frac{FR_p N_{F_p}}{M_c}$$

$$-\left[\lambda_p + D_p + \frac{Q_L}{M_c}\left(\frac{DF_p - 1}{DF_p}\right)\right] N_{c_p}$$

For daughter nuclides in the coolant:

$$\frac{dN_{c_d}}{dt} = \frac{FR_d N_{F_d}}{M_c} + f_p\, \lambda_p\, N_{c_p} -$$

$$\left[\lambda_d + D_d + \frac{Q_L}{M_c}\left(\frac{DF_d - 1}{DF_d}\right)\right] N_{c_d}$$

where:

N_c	=	Concentration of nuclide in the reactor coolant (atoms/gram)
N_F	=	Population of nuclide in the fuel (atoms)
t	=	Operating time (seconds)
R	=	Nuclide release coefficient (1/sec)
F	=	Fraction of fuel rods with defective cladding
M_c	=	Mass of reactor coolant (grams)
λ	=	Nuclide decay constant (1/sec)
D	=	Dilution coefficient by feed and bleed (1/sec) $= \frac{\beta}{B_o - \beta t} \times$ [illegible]
B_o	=	Initial boron concentration (ppm)
β	=	Boron concentration reduction rate (ppm/sec)
DF	=	Nuclide demineralizer decontamination factor
Q_L	=	Purification or letdown mass flow rate (grams/sec)
f	=	Fraction of parent nuclide decay events that result in the formation of the daughter nuclide

Subscript p refers to the parent nuclide.

Subscript d refers to the daughter nuclide.

Table 11.1-2 lists the resulting reactor coolant radionuclide concentrations. The values presented are the maximum values calculated to occur during the fuel cycle from startup through the equilibrium cycle. Thus, the source term does not represent any particular time in the fuel cycle but is a conservative composite.

The design basis source term based on 0.25 percent fuel defects is used to ensure a consistent set of design values for interfaces among the radioactive waste processing systems. The Technical Specifications in Chapter 16, which are related to fuel failure are also based upon 0.25 percent fuel defects. In addition, the liquid and gaseous radioactive waste processing systems have the capability to process wastes based upon 1.0 percent fuel defects.

11.1.1.2 Corrosion Products

The reactor coolant corrosion product activities are based on operating plant data and are independent of fuel defect level. The concentrations of corrosion products are included in Table 11.1-2.

11.1.1.3 Tritium

A number of tritium production processes add tritium to the reactor coolant:

- Fission product formation in the fuel (ternary fission) forms tritium which can diffuse through the fuel clad or leak through fuel clad defects
- Neutron reactions with soluble boron in the reactor coolant
- Burnable neutron absorber
- Neutron reactions with soluble lithium in the reactor coolant
- Neutron reactions with deuterium in the reactor coolant

The first two processes are the principal contributors to tritium in the reactor coolant. Table 11.1-3 lists the tritium introduced to the reactor coolant from each of the processes.

Tritium exists in the reactor coolant primarily as tritium oxide (that is, a tritium atom replaces a hydrogen atom in a water molecule) and thus cannot be readily separated from the coolant by normal processing methods. The maximum concentration of tritium in the reactor coolant is less than 3.5 microcuries per gram as a result of losses due to leakage and the controlled release of tritiated water to the environment.

11.1.1.4 Nitrogen-16

Activation of oxygen in the coolant results in the formation of N-16 which is a strong gamma emitter. Because of its short half-life of 7.11 seconds, N-16 is not of concern outside the containment. Table 12.2-3 provides N-16 concentrations at various points in the reactor coolant system. After shutdown, N-16 is not a source of radiation inside of containment.

11.1.2 Design Basis Secondary Coolant Activity

Steam generator tube defects cause the introduction of reactor coolant into the secondary cooling system. The resulting radionuclide concentrations in the secondary coolant depend upon the primary-to-secondary leak rate, the nuclide decay constant, and the steam generator blowdown rate.

The reactor coolant leakage into the secondary system is assumed to have radionuclide concentrations as defined in Table 11.1-2. The parameters used in the calculation of the secondary side activities are provided in Table 11.1-4 and the resulting radionuclide concentrations in the steam generator secondary side water and steam are presented in Tables 11.1-5 and 11.1-6.

11.1.3 Realistic Reactor Coolant and Secondary Coolant Activity

The realistic source terms for both the reactor coolant and the secondary coolant are determined using the modeling in ANSI-18.1 (Reference 3). This modeling is also incorporated in the PWR-GALE code. The reference plant values provided in ANSI-18.1 were adjusted to be consistent with the AP600 parameters listed in Table 11.1-7. The adjustment factors are applied to the fission products. The realistic source terms are listed in Table 11.1-8.

11.1.4 Core Source Term

The core fission product inventories used to establish source terms for accident radiological consequence analyses are provided in Appendix 15A.

11.1.5 Process Leakage Sources

The systems containing radioactive liquids are potential sources for the release of radioactive material to plant buildings and then to the environment. The leakage sources and the resulting airborne concentrations are discussed in Section 12.2.

Release pathways for radioactive materials are discussed in Sections 11.2 and 11.3.

11.1.6 Combined License Information

This section has no requirement for information to be provided in support of the Combined License application.

11.1.7 References

1. "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," NUREG-0017, Revision 1, March 1985.

2. RSIC Computer Code Collection CCC-371, ORIGEN 2.1 Isotope Generation and Depletion Code - Matrix Exponential Method, August 1, 1991.

3. ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors."

Table 11.1-1 (Sheet 1 of 2)

PARAMETERS USED IN THE CALCULATION OF DESIGN BASIS FISSION PRODUCT ACTIVITIES

Core thermal power (MWt)	1,933
Reactor coolant liquid volume (ft^3)	7,654
Reactor coolant full-power average temperature (°F)	561.5
Purification flow rate (gal/min)[(a)]	100
Effective cation demineralizer flow, annual average (gal/min)[(a)]	10
Nuclide release coefficients (the product of the failed fuel fraction and the fission product escape rate coefficient)	
Equivalent fraction of core power produced by fuel rods containing small cladding defects (failed fuel fraction)	0.0025
Fission product escape rate coefficients during full-power operation (s^{-1}):	
Kr and Xe isotopes	6.5×10^{-8}
Br, Rb, I, and Cs isotopes	1.3×10^{-8}
Mo, Tc, and Ag isotopes	2.0×10^{-9}
Te isotopes	1.0×10^{-9}
Sr and Ba isotopes	1.0×10^{-11}
Y, Zr, Nb, Ru, Rh, La, Ce, and Pr isotopes	1.6×10^{-12}
Chemical and volume control system mixed bed demineralizers	
Resin volume (ft^3)	50
Demineralizer isotopic decontamination factors:	
Kr and Xe isotopes	1
Br and I isotopes	10
Sr and Ba isotopes	10
Other isotopes	1

Table 11.1-1 (Sheet 2 of 2)

PARAMETERS USED IN THE CALCULATION OF DESIGN BASIS FISSION PRODUCT ACTIVITIES

Chemical and volume control system cation bed demineralizer	
Resin volume (ft^3)	50
Demineralizer isotopic decontamination factors:	
Kr and Xe isotopes	1
Sr and Ba isotopes	1
Rb-86, Cs-134, and Cs-137	10
Rb-88, Rb-89, Cs-136, and Cs-138	1
Other isotopes	1
Other isotopic removal mechanisms	See Note b
Initial boron concentration (ppm)	1,753
Operation time (effective full-power hours)	15,250

Notes:

a. Flow calculated at 2250 psia and 130°F.

b. For all isotopes, except the isotopes of Kr, Xe, Br, I, Rb, Cs, Sr, and Ba, a removal decontamination factor of 10 is assumed to account for removal mechanisms other than ion exchange, such as plateout. This decontamination factor is applied to the full purification flow.

Table 11.1-2

DESIGN BASIS REACTOR COOLANT ACTIVITY

Nuclide	Activity (μCi/g)	Nuclide	Activity (μCi/g)
Kr-83m	8.9×10^{-2}	Rb-89	3.5×10^{-2}
Kr-85m	4.7×10^{-1}	Rb-88	7.6×10^{-1}
Kr-85	2.1	Sr-89	4.1×10^{-4}
Kr-87	2.5×10^{-1}	Sr-90	2.5×10^{-5}
Kr-88	7.7×10^{-1}	Sr-91	9.8×10^{-4}
Kr-89	1.7×10^{-2}	Sr-92	2.0×10^{-4}
Xe-131m	7.5×10^{-1}	Y-90	6.5×10^{-6}
Xe-133m	9.8×10^{-1}	Y-91m	5.2×10^{-4}
Xe-133	$6.9 \times 10^{+1}$	Y-91	5.4×10^{-5}
Xe-135m	7.5×10^{-2}	Y-92	1.8×10^{-4}
Xe-135	2.1	Y-93	4.9×10^{-5}
Xe-137	3.4×10^{-2}	Zr-95	6.4×10^{-5}
Xe-138	1.2×10^{-1}	Nb-95	6.4×10^{-5}
Br-83	1.5×10^{-2}	Mo-99	7.8×10^{-2}
Br-84	7.9×10^{-3}	Tc-99m	7.2×10^{-2}
Br-85	9.6×10^{-4}	Ru-103	5.5×10^{-5}
I-129	6.7×10^{-9}	Rh-103m	5.7×10^{-5}
I-130	4.0×10^{-3}	Rh-106	2.0×10^{-5}
I-131	2.8×10^{-1}	Ag-110m	1.8×10^{-4}
I-132	4.8×10^{-1}	Te-129m	1.0×10^{-3}
I-133	5.8×10^{-1}	Te-127m	3.1×10^{-4}
I-134	1.0×10^{-1}	Te-129	1.6×10^{-3}
I-135	3.2×10^{-1}	Te-131m	2.7×10^{-3}
Cs-134	2.3×10^{-1}	Te-131	2.0×10^{-3}
Cs-136	5.3×10^{-1}	Te-132	3.1×10^{-2}
Cs-137	2.1×10^{-1}	Te-134	5.0×10^{-3}
Cs-138	1.8×10^{-1}	Ba-137m	2.0×10^{-1}
Cr-51	1.3×10^{-3}	Ba-140	4.1×10^{-4}
Mn-54	6.7×10^{-4}	La-140	1.1×10^{-4}
Mn-56	1.7×10^{-1}	Ce-141	6.4×10^{-5}
Fe-55	5.0×10^{-4}	Ce-143	5.3×10^{-5}
Fe-59	1.3×10^{-4}	Pr-143	5.9×10^{-5}
Co-58	1.9×10^{-3}	Ce-144	4.8×10^{-5}
Co-60	2.2×10^{-4}	Pr-144	4.8×10^{-5}

Note:
These activities are used for shielding and radwaste system interface design. For 1 percent fuel defect calculations (maximum release and liquid and gaseous radwaste system capability) multiply the activities above by 4 except for iodine, noble gases and corrosion products.

Table 11.1-3

TRITIUM SOURCES

Tritium Source	Release to the Coolant (curies/cycle[1]) Design Basis	Best Estimate
Produced in the core		
Ternary fission	1241	249
Burnable absorbers	87	17
Produced in the coolant		
Soluble boron	371	371
Soluble lithium	95	95
Deuterium	2	2
TOTAL	1796	734

[1] Cycle length of 24 months. Design basis case reflects the historical assumption that 10% of the tritium produced in the core is released to the coolant. Best estimate case is based on a release of only 2% of the tritium.

Table 11.1-4

PARAMETERS USED TO CALCULATE SECONDARY COOLANT ACTIVITY

Total secondary side water mass (lb)	2.12×10^5
Steam generator steam fraction	0.055
Total steam flow rate (lb/hr)	8.4×10^6
Moisture carryover (percent)	0.1
Total makeup water feed rate (lb/hr)	442
Total blowdown rate (lb/hr)	8.5×10^4
Total primary-to-secondary leak rate (gpd)	500
Iodine partition factor (mass basis)	100

Table 11.1-5

DESIGN BASIS STEAM GENERATOR SECONDARY SIDE LIQUID ACTIVITY

Nuclide	Activity (μCi/g)	Nuclide	Activity (μCi/g)
Br-83	1.4×10^{-5}	Y-93	6.9×10^{-7}
Br-84	2.9×10^{-6}	Zr-95	1.1×10^{-7}
Br-85	3.9×10^{-8}	Nb-95	1.1×10^{-7}
I-129	1.1×10^{-11}	Mo-99	1.3×10^{-4}
I-130	5.8×10^{-6}	Tc-99m	1.2×10^{-4}
I-131	4.6×10^{-4}	Ru-103	9.2×10^{-8}
I-132	4.6×10^{-4}	Rh-103m	9.3×10^{-8}
I-133	8.8×10^{-4}	Rh-106	1.4×10^{-10}
I-134	5.4×10^{-5}	Ag-110m	3.0×10^{-7}
I-135	4.1×10^{-4}	Te-127m	5.2×10^{-7}
Rb-88	1.8×10^{-4}	Te-127	8.8×10^{-8}
Rb-89	7.2×10^{-6}	Te-129m	1.7×10^{-6}
Cs-134	6.4×10^{-4}	Te-131m	4.3×10^{-6}
Cs-136	1.6×10^{-3}	Te-131	1.2×10^{-6}
Cs-137	6.3×10^{-4}	Te-132	5.1×10^{-5}
Ba-137m	5.9×10^{-4}	Te-134	2.2×10^{-6}
Cs-138	7.3×10^{-5}	Ba-140	6.8×10^{-6}
H-3	1.0	La-140	1.8×10^{-7}
Cr-51	2.2×10^{-6}	Ce-141	1.1×10^{-7}
Mn-54	1.1×10^{-6}	Ce-143	8.4×10^{-8}
Mn-56	1.6×10^{-4}	Ce-144	8.0×10^{-8}
Fe-55	8.4×10^{-7}	Pr-143	9.9×10^{-8}
Fe-59	2.2×10^{-7}	Pr-144	8.0×10^{-8}
Co-58	3.2×10^{-6}		
Co-60	3.7×10^{-7}		
Sr-89	1.3×10^{-6}		
Sr-90	7.5×10^{-8}		
Sr-91	2.2×10^{-6}		
Sr-92	2.7×10^{-7}		
Y-90	1.3×10^{-8}		
Y-91m	1.1×10^{-6}		
Y-91	9.3×10^{-8}		
Y-92	2.0×10^{-7}		

Table 11.1-6

DESIGN BASIS STEAM GENERATOR SECONDARY SIDE STEAM ACTIVITY

Nuclide	Activity (μCi/g)
Kr-83m	1.5×10^{-6}
Kr-85m	7.2×10^{-6}
Kr-85	3.2×10^{-5}
Kr-87	3.8×10^{-6}
Kr-88	1.2×10^{-5}
Kr-89	2.5×10^{-7}
Xe-131m	1.1×10^{-5}
Xe-133m	1.5×10^{-5}
Xe-133	1.1×10^{-3}
Xe-135m	4.2×10^{-6}
Xe-135	3.3×10^{-5}
Xe-137	5.1×10^{-7}
Xe-138	1.8×10^{-6}
I-129	1.2×10^{-13}
I-130	6.3×10^{-8}
I-131	5.1×10^{-6}
I-132	5.0×10^{-6}
I-133	9.7×10^{-6}
I-134	5.9×10^{-7}
I-135	4.5×10^{-6}
H-3	1.0

Table 11.1-7

PARAMETERS USED TO DESCRIBE REALISTIC SOURCES

Parameter	Symbol	Units	AP600 Value	Nominal Value
Thermal power	P	MWt	1933	3400
Steam flow rate	FS	lb/hr	8.4×10^6	1.5×10^7
Weight of water in reactor coolant system	WP	lb	3.5×10^5	5.5×10^5
Weight of water in all steam generators	WS	lb	2.1×10^5	4.5×10^5
Reactor coolant purification flow	FD	lb/hr	5.0×10^4	3.7×10^4
Reactor coolant letdown flow (yearly average for boron control)	FB	lb/hr	1.2×10^2	5.0×10^2
Steam generator blowdown flow (total)	FBD	lb/hr	4.2×10^4	7.5×10^4
Fraction of radioactivity in blowdown stream which is not returned to the secondary coolant system	NBD	-	0.0	1.0
Flow through the purification system cation demineralizer	FA	lb/hr	5.0×10^3	3.7×10^3
Ratio of condensate demineralizer flow rate to the total steam flow rate	NC	-	0.33	0.0
Fraction of the noble gas activity in the letdown stream which is not returned to the reactor coolant system	Y	-	0.0	0.0
Primary-to-secondary leakage	FL	lb/day	75	75

Table 11.1-8 (Sheet 1 of 4)

REALISTIC SOURCE TERMS

Noble Gases

Nuclide	Reactor Coolant Activity (µCi/g)	Steam Generator Steam Activity (µCi/g)
Kr-85m	0.14	5.4×10^{-8}
Kr-85	0.96	3.6×10^{-7}
Kr-87	0.13	4.8×10^{-8}
Kr-88	0.25	9.4×10^{-8}
Xe-131m	0.78	2.9×10^{-7}
Xe-133m	0.065	2.5×10^{-8}
Xe-133	2.5	9.4×10^{-7}
Xe-135m	0.12	4.3×10^{-8}
Xe-135	0.76	2.9×10^{-7}
Xe-137	0.030	1.1×10^{-8}
Xe-138	0.11	4.0×10^{-8}

Halogens

Nuclide	Reactor Coolant Activity (µCi/g)	Steam Generator Liquid Activity (µCi/g)	Steam Generator Steam Activity (µCi/g)
Br-84	0.014	1.4×10^{-7}	1.4×10^{-9}
I-131	0.02	2.3×10^{-6}	2.3×10^{-8}
I-132	0.16	5.4×10^{-6}	5.4×10^{-8}
I-133	0.073	6.9×10^{-6}	6.9×10^{-8}
I-134	0.28	4.3×10^{-6}	4.3×10^{-8}
I-135	0.16	1.1×10^{-5}	1.1×10^{-7}

Table 11.1-8 (Sheet 2 of 4)

REALISTIC SOURCE TERMS

Rubidium, Cesium

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Generator Liquid Activity (μCi/g)	Steam Generator Steam Activity (μCi/g)
Rb-88	0.17	1.0×10^{-6}	4.9×10^{-9}
Cs-134	2.9×10^{-3}	1.3×10^{-6}	6.7×10^{-9}
Cs-136	3.7×10^{-4}	1.5×10^{-7}	7.7×10^{-10}
Cs-137	3.9×10^{-3}	1.7×10^{-6}	8.6×10^{-9}

Tritium

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Generator Liquid Activity (μCi/g)	Steam Generator Steam Activity (μCi/g)
H-3	1	1.0×10^{-3}	1.0×10^{-3}

Table 11.1-8 (Sheet 3 of 4)

REALISTIC SOURCE TERMS

Miscellaneous Nuclides

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Generator Liquid Activity (μCi/g)	Steam Generator Steam Activity (μCi/g)
Na-24	2.5×10^{-2}	3.4×10^{-6}	1.7×10^{-8}
Cr-51	1.3×10^{-3}	3.2×10^{-7}	1.5×10^{-9}
Mn-54	6.6×10^{-4}	1.6×10^{-7}	8.1×10^{-10}
Fe-55	5.0×10^{-4}	1.2×10^{-7}	6.1×10^{-10}
Fe-59	1.2×10^{-4}	2.9×10^{-8}	1.5×10^{-10}
Co-58	1.9×10^{-3}	4.6×10^{-7}	2.3×10^{-9}
Co-60	2.2×10^{-4}	5.4×10^{-8}	2.7×10^{-10}
Zn-65	2.1×10^{-4}	5.1×10^{-8}	2.4×10^{-10}
Sr-89	5.8×10^{-5}	1.4×10^{-8}	7.1×10^{-11}
Sr-90	5.0×10^{-6}	1.2×10^{-9}	6.1×10^{-12}
Sr-91	5.5×10^{-4}	6.1×10^{-8}	3.1×10^{-10}
Y-91m	3.8×10^{-4}	6.1×10^{-9}	3.1×10^{-11}
Y-91	2.2×10^{-6}	5.1×10^{-10}	2.7×10^{-12}
Y-93	2.4×10^{-3}	2.7×10^{-7}	1.3×10^{-9}
Zr-95	1.6×10^{-4}	3.9×10^{-8}	1.9×10^{-10}
Nb-95	1.2×10^{-4}	2.7×10^{-8}	1.4×10^{-10}
Mo-99	2.9×10^{-3}	6.0×10^{-7}	2.9×10^{-9}
Tc-99m	3.0×10^{-3}	2.3×10^{-7}	1.2×10^{-9}
Ru-103	3.1×10^{-3}	7.6×10^{-7}	3.9×10^{-9}
Ru-106	3.7×10^{-2}	9.0×10^{-5}	4.4×10^{-8}
Rh-103m	6.1×10^{-3}	6.0×10^{-7}	3.1×10^{-9}
Rh-106	8.0×10^{-2}	6.8×10^{-5}	3.3×10^{-8}
Ag-110m	5.4×10^{-4}	1.3×10^{-7}	6.6×10^{-10}
Te-129m	7.9×10^{-5}	1.9×10^{-8}	9.5×10^{-11}

Table 11.1-8 (Sheet 4 of 4)

REALISTIC SOURCE TERMS

Miscellaneous Nuclides

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Generator Liquid Activity (μCi/g)	Steam Generator Steam Activity (μCi/g)
Te-129	1.9×10^{-2}	4.3×10^{-7}	2.1×10^{-9}
Te-131m	7.1×10^{-4}	1.3×10^{-7}	6.3×10^{-10}
Te-131	6.6×10^{-3}	5.5×10^{-8}	2.8×10^{-10}
Te-132	7.5×10^{-4}	1.6×10^{-7}	7.9×10^{-10}
Ba-137m	3.7×10^{-3}	1.6×10^{-6}	8.2×10^{-9}
Ba-140	5.5×10^{-3}	1.3×10^{-6}	6.3×10^{-9}
La-140	1.2×10^{-2}	2.2×10^{-6}	1.1×10^{-8}
Ce-141	6.2×10^{-5}	1.5×10^{-8}	7.6×10^{-11}
Ce-143	1.3×10^{-3}	2.3×10^{-7}	1.2×10^{-9}
Ce-144	1.7×10^{-3}	3.9×10^{-7}	2.0×10^{-9}
Pr-143	1.5×10^{-3}	2.9×10^{-7}	1.6×10^{-9}
Pr-144	3.4×10^{-3}	3.0×10^{-7}	1.5×10^{-9}
W-187	1.2×10^{-3}	2.0×10^{-7}	1.0×10^{-9}
Np-239	9.9×10^{-4}	2.0×10^{-7}	1.0×10^{-9}