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00	First issue.	28-04-08
01	Corrections of errors in Tables 1 and 2 - Clarifications	25-11-08
02	PCSR June 2009 update: - Inclusion of references; - Clarification of text - Erratum correction in figures Update equivalent PCER chapter	19-09-09
03	PCER March 2010 update - Relevant references marked as CCI - Complementary information included in section 2.8	21-03-10
04	PCER March 2011 update: - Minor editorial changes - Update of references - Text added to specify the use of each source term and reference added (§1)	29-03-11
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05	Consolidated PCER update: - References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc - Minor formatting changes	21-08-12

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REVISION HISTORY (Cont'd)

Issue	Description	Date
05 (cont'd)	- Minor editorial changes (§1 and §2.1) - Clarification of text (introductory text, §2.6, §2.7, Table 2) - Update to reactor chemistry aspects, for zinc injection and source term, and references added (§2.5, §2.6, §2.7)	21-08-12

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SUB-CHAPTER 6.1 - SOURCES OF RADIOACTIVE MATERIALS

LIST OF TABLES

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This sub-chapter deals with requirement 2.1 of EA P&I Document [Ref-1].

The radioactive products likely to be discharged are produced in the core.

In the primary coolant, they are present in the following forms:

- fission products, likely to be released through small defects in the fuel rod cladding during unit operation;
- corrosion products released by the primary system internal structures and activated as they pass into the core active zone;
- primary coolant activation products such as ^{3}H (tritium) and ^{16}N (nitrogen).

The level of radioactivity in the primary coolant and the connected systems is used to assess the radiological consequences of accidents without deterioration of the core, radiological protection, and the sizing of nuclear buildings (biological shielding).

The activity levels of the main primary system when the reactor is in normal operation (steady-state operation and shutdown transient) have been determined using three source terms: the realistic source term, the biological protection design source term and the effluent treatment system design source term. In addition, a single source term is determined in Sub-chapter 12.2 of the PCSR for the activities deposited on the wall of pipes. These source terms have been used in Chapter 12 of the PCSR for the radiological protection calculations. The activity values determined may also be used in a generic way as interface data for all other chapters dealing with effluent discharge, radiological protection and assessment of realistic dose uptake rates, both inside and outside the containment, covering normal conditions, waste management and accident analysis.

The kinetics increase of activity in the primary coolant (usually described as ‘fission product spiking’) after unit shutdown is of particular interest for the evaluation of the radiological consequences of PCC-2 to PCC-4 events, especially for long-lived radio-nuclides. This increase is taken into account in the calculations of the radiological consequences of accidents without additional clad failure (see Sub-chapter 14.6 of the PCSR).

1. DEFINITIONS OF THE SOURCE TERMS USED

For the primary coolant, three types of activity values have been selected to characterise normal operating conditions [Ref-1]:

- **Realistic source term:**

The realistic source term, historically estimated based on operational experience feedback from German and French units, represents the average specific activities most likely to be seen during normal operating conditions. This source term is representative of the average values that may be measured in the primary coolant water. This source term encompasses the average values measured on the N4 series, and is proposed for phases corresponding to both steady-state operation and shutdown transient (in particular during reduction of load for fission products and during the oxygenation peak for corrosion products).

Sub-chapter 6.1 – Table 1 specifies the realistic values expected during steady-state operation of the reactor.

Sub-chapter 6.1 – Table 2 specifies the realistic activity values expected during shutdown transients (at the time of reduction of load for fission products, and the oxygenation peak for corrosion products).

The realistic source term was initially defined for:

- the French ESPN classification[1];
- worker dose assessment. In the context of PCSR Sub-chapter 12.4, the collective dose was determined on the basis of the statistics for dose uptakes on the best performing French power plants. The EPR design optimisation was taken into account in evaluating both additional doses and dose improvements (due to EPR design modifications), and to calculate the EPR collective dose. For conservatism, the realistic source term was not used in the final calculation of the workers' dose. [Ref-2].

- **Biological protection design source term (DPB source term):**

The source term is more conservative than the realistic source term. It corresponds to specific activity values covering all spectrometry measurements obtained on the N4 series.

Sub-chapter 6.1 – Table 1 specifies the activity values corresponding to the biological protection design source term expected during steady-state operation of the reactor.

Sub-chapter 6.1 – Table 2 specifies the activity values corresponding to the biological protection design source term expected during shutdown transients (at the time of reduction of load for fission products, and the oxygenation peak for corrosion products).

This source term was defined for the design and sizing of the structures, rooms, systems and shielding of the EPR [Ref-2].

- **Effluent treatment system design source term (DSE source term):**

This source term is more conservative than the realistic and the DPB source terms.

For fission products, activity values are normalised with the radiochemical specifications of existing plants (equivalent iodine-131 of 20 GBq/t in steady-state operation and 150 GBq/t in a power transient) and cover all of the spectrometry measurements obtained on 1300 MWe and N4 series such that the possibility of fuel clad ruptures are taken into account.

For corrosion products, the activity values selected correspond to the maximum values measured by spectrometry on the N4 series. Feedback from the 1300 MWe series is not taken into account owing to the differences in the design materials of the steam generator tubes (mostly alloy 600 on the 1300 MWe series compared with alloy 690 on the N4 series). The effluent treatment system design source term is proposed for phases corresponding to steady-state operation and for shutdown transients (at the time of load reduction for fission products and the oxygenation peak for corrosion products).

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[1] The French ESPN (Nuclear Pressure Equipment) regulatory classification has no equivalent in the British regulations.

Sub-chapter 6.1 – Table 1 specifies the activity values corresponding to the effluent treatment system design source term expected during steady-state operation of the reactor.

Sub-chapter 6.1 – Table 2 specifies the activity values corresponding to the effluent treatment system design source term expected during shutdown transients (at the time of reduction of load for fission products, and the oxygenation peak for corrosion products).

The DSE source term was initially defined for [Ref-2]:

- sizing the effluent treatment systems;
- performing the radiological consequences studies.

2. ISOTOPE INVENTORY

2.1. NITROGEN-16

Nitrogen-16 (^{16}N) is formed by the activation of oxygen-16 (moderator water molecule) by fast neutrons over the entire energy spectrum. This is the most important nuclide in the primary system from a radiation point of view. The concentration of activity depends mainly on the power level (neutron flux) and the duration of the passage of the water through the core (derived from geometric data). It has a half-life of 7.3 seconds. It is calculated using the following formula for successive disintegrations and decays [Ref-1]:

$$A_n = N\sigma\Phi.\left(\frac{1-e^{-\lambda t}}{1-e^{-\lambda\tau}}\right).\left[1-e^{-(n-1)\lambda\tau}\right]$$

where:
n: disintegration number;
An: specific activity after the nth disintegration (Bq/t);
N: number of target nuclides (^{16}O) per tonne of water in the RCP [RCS];
σ : effective cross-section for the ^{16}O (n, p) ^{16}N reaction, averaged over the fission spectrum;
Φ : neutron flux (energy > 1 MeV);
λ : radioactive decay constant of ^{16}N;
t: radiation duration;
τ : total transit time in the primary loop (radiation plus decay periods).

Nitrogen-16 is a powerful γ emitter and thus forms the main radioactive source for external exposure of a worker during operation of the reactor. The concentration of nitrogen-16 in the primary circuit is given in Sub-chapter 6.1 – Figure 1.

2.2. NITROGEN-17

Nitrogen-17 (^{17}N) is also an activation product, which comes from the reaction of neutrons on the oxygen-17 in the primary coolant. The nitrogen-17 decays in a few seconds (half-life of 4.2 seconds) to an excited state of the oxygen-17, which tends to emit neutrons. This nuclide may thus be an additional source of neutrons (with respect to those from the core) to be taken into account for operator accessibility in the Reactor Building (operation at full power).

The concentration is given in Sub-chapter 6.1 – Figure 2. It has been calculated using the same methodology as that described above for nitrogen-16.

2.3. TRITIUM

Tritium is produced both by fission reactions and by neutron activation of boron (mostly B-10), lithium (mostly Li-6) or deuterium.

The actual concentration in the primary coolant depends on the liquid effluent treatment policy (recycling versus release) and the initial boron concentration in the primary coolant. Since tritium is a β pure emitter it is not involved in the sizing of biological protection; a single all-purpose value has been set at 3.7×10^{10} Bq/t (see Sub-chapter 6.1 - Table 1) [Ref-1]. This value has been used to determine internal exposures due to atmospheric radioactivity (see Sub-chapter 12.4 of the PCSR). The atmospheric tritium content is not given as a limiting value.

Tritium is also taken into account in the assessment of the atmospheric activity in the Reactor Building and the Fuel Building. The level depends on the humidity in the air, the atmospheric recirculation rate, the tritium content in the sump and in the Reactor Building, and the leak rate from the primary system.

The values of the above reference table are for guidance only. They may be modified at a later date depending on the legal requirements and the limits for gaseous and liquid effluent specific to the site and criteria from regulators concerning the maximum tritium content in the RCP [RCS]. In the EDF fleet units, the target content is based on a compromise between waste management and dose limitation.

2.4. ARGON-41

This nuclide is the result of activation of natural Argon-40 in the air that dissolves in the water of the primary circuit during cold shutdown operations. Although processes are in place during start-up to eliminate air as much as possible from the primary circuit (in particular oxygen for safety reasons), there remains some Ar-40 dissolved in water, which is likely to be activated by neutronic flux. For this reason, its production rate is directly linked to the neutron flux in this region and thus to the power level. Production is mostly observed at the beginning of the irradiation cycles[2]. Ar-41 decays with a half-life of 1.8 hours, also emitting gamma rays.

2.5. CARBON-14

The half life of carbon-14 is 5,730 years. For this reason, this β emitter is taken into account for atmospheric radioactivity and gaseous releases. Its main production sources are:

- (n, p) reaction with nitrogen-14 (air around the reactor vessel and air in the containment);
- (n, α) reaction with oxygen-17 (primary coolant);

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[2] Due to the open structure and the ventilation requirements in the reactor building, Argon-41 may also be responsible for external gamma exposure when individuals enter the containment while the reactor is in operation. The values provided in this sub-chapter however only concern the primary circuit and do not provide any activation value of the air close to the reactor vessel.

- fission reaction.

Carbon-14 is also produced from carbon activation but this production is very low compared to that formed by oxygen and nitrogen, even with zinc injection [Ref-1].

The realistic value used for the EPR primary coolant is 6 MBq/t, with an upper bound value of 13 MBq/t. These two values have been estimated for the EPR based on measurements taken from the primary water in the 1300 MWe and 900 MWe units, taking into account the difference between the EPR purification rate and that of the 1300 and 900 MWe units during operation [Ref-2].

2.6. CORROSION PRODUCTS

The specific concentrations and the shutdown spiking factors of radio-nuclides present in the main primary system are based on the measurements taken from French N4 units. The radio-nuclides considered are shown in the following table:

Radionuclide	Radioactive half life	Origin
Mn-54	312.5 days	All metallic materials
Co-58	70.78 days	Nickel-based alloys (Inconel 690)
Fe-59	45.1 days	All metallic materials
Co-60	5.27 years	Stellites, impurities of other metallic materials
Cr-51	27.7 days	Activation of corrosion products released by stainless steels and 690 nickel-based alloy
Ni-63	100 years	690 nickel-based alloys
Ag-110m	249.9 days	Control rods in AIC Buttering of seals (helicoflex)
Sb-122	2.7 days	Activation of Sb-121 contained in antimony-based alloys used for the bearings for some pumps Secondary source rod breakages
Sb-124	60.2 days	Activation of Sb-123 contained in antimony-based alloys used for the bearings for some pumps Secondary source rod breakages
Sb-125	2.73 years	Activation of unstable Sb-124

The corresponding values (see Sub-chapter 6.1 – Table 1 for steady-state operation and Sub-chapter 6.1 – Table 2 for shutdown transients) take into account an average release of materials for a major part of unit operation, as well as the improvement in the manufacturing of the steam generator tubes. Furthermore, adequate surface treatment of parts of the RCP [RCS] and appropriate chemical specification also reduce the production rate of corrosion products but these have not been taken into account since the associated gains are difficult to evaluate.

Moreover, in order to avoid excessive contamination levels by silver-110m, antimony-124 and antimony-122, the design of the primary components in contact with the primary coolant aims to avoid as far as possible the source metals of the responsible nuclides.

In this context, the following improvements are already planned:

- reduction in the use of helicoflex seals in favour of graphite seals;
- a greater use of bearings without antimony;

- use of mechanical seals without antimony on applicable pumps.

Nevertheless, these improvements have not been taken into account in the evaluation of the corrosion products source terms due to the difficulty of quantification of the reduction brought by these improvements.

Quantification of the RCP [RCS] source term, based on calculations and considering specific materials and chemistry conditions of UK EPR [Ref-1] [Ref-2], shows the consistency of the nuclide source term specified in this sub-chapter.

2.7. FISSION PRODUCTS

The radio-nuclides considered in the EPR studies are as follows:

- *noble gases:* Kr-85m (4.48 hours), Kr-85 (10.72 years), Kr-87 (1.27 hours), Kr-88 (2.84 hours), Xe-131m (11.9 days), Xe-133m (2.19 days), Xe-133 (5.25 days), Xe-135 (9.09 hours), Xe-138 (14.2 minutes);
- *strontium*: Sr-89 (50.5 days), Sr-90 (29.2 years);
- *iodine*: I-131 (8.04 days), I-132 (2.3 hours), I-133 (20.8 hours), I-134 (52.6 minutes), I-135 (6.61 hours);
- *caesium*: Cs-134 (2.06 years), Cs-136 (13.16 days), Cs-137 (30 years), Cs-138 (32.2 minutes).

The source term values are presented in Sub-chapter 6.1 - Table 1 for steady-state operation and Sub-chapter 6.1 - Table 2 for shutdown transients.

Quantification of the RCP [RCS] source term, based on calculations and considering specific materials and chemistry conditions of UK EPR [Ref-1] [Ref-2], shows the consistency of the nuclide source term specified in this sub-chapter.

2.8. ACTINIDES

Most of the actinides produced in a PWR result from the neutronic activation of uranium. In the absence of fuel cladding defects, actinides arising in the primary circuit might result from two sources:

- traces of uranium on the outside of the cladding left over from manufacture of the fuel;
- impurities in the fuel cladding and the other material.

However, these two sources are not significant compared to the quantity of uranium potentially released in the primary circuit in the event of fuel cladding defects.

The presence of fissile material in the primary circuit leads to two processes of contamination:

- dissemination of alpha emitters in the primary water and their deposit on the internal surfaces of circuits;
- creation of fission products.

Even on sites that experienced fuel cladding defects, global alpha activity measured both in liquid and gaseous effluents has always been lower than the limits of detection, due to the high efficiency of filtration.

It is important to note that improvement of the fuel reliability is a major objective for the EPR. A worldwide program including manufacturing, human aspects, research and development has been developed. The EPR fuel (i.e. AFA 3GLE) includes in its design and benefits from, on the manufacturing and quality process fronts as well, all the improvements that are the results of years of research and development.

The annual fuel rod failure rate is a recognised indicator of the operational reliability of fuel assemblies. It is determined as the ratio of number of failed rods discharged divided by the number of fuel rods in reactors which have been refuelled during the considered year. AREVA's PWR fuel assemblies have exhibited consistently high operational reliability with an average annual fuel failure rate of approximately 10^{-5}. Over the past five years the failure rate has been reduced to less than half of the failure rate at the end of the 1980s due to AREVA’s ongoing effort to increase fuel reliability whilst reactor operating conditions become increasingly demanding.

SUB-CHAPTER 6.1 - TABLE 1

Specific Nuclide Concentrations in the Primary System (RCP [RCS] 1, 2, 3, 4) - Stabilised Operation [Ref-1]

NUCLIDE	SPECIFIC ACTIVITY (MBq/t)		
	Realistic	Biological protection design	Effluent treatment system design
Mn-54	4.2	220	220
Co-58	21	390	390
Fe-59	1.3	81	81
Co-60	2.3	170	170
Cr-51	28	600	600
Ag-110m	3.2	270	270
Sb-122	1,2	110	110
Sb-124	0.97	120	120
Sb-125	11	98	98
Ni-63	15	15	15
Kr-85m	200	5500	15000
Kr-85	38	620	2400
Kr-87	360	10000	23000
Kr-88	500	14000	35000
Xe-131m	28	440	1700
Xe-133m	110	1700	11000
Xe-133	5000	80000	310000
Xe-135	1100	18000	92000
Xe-138	850	14000	72000
I-131	100	1600	15000
I-132	190	2800	18000
I-133	310	4900	24000
I-134	190	1800	7700
I-135	200	3300	16000
Cs-134	40	320	4500
Cs-136	3.7	33	2100
Cs-137	40	320	3300
Cs-138	850	14000	100000
Sr-89	0.3	4.9	30
Sr-90	0.0019	0.03	0.19
N-16	*see Sub-chapter 6.1 - Figure 1*		
N-17	*see Sub-chapter 6.1 - Figure 2*		
H-3	37000	37000	37000
Ar-41	300	1000	3000
C-14	6	13	13

SUB-CHAPTER 6.1 - TABLE 2

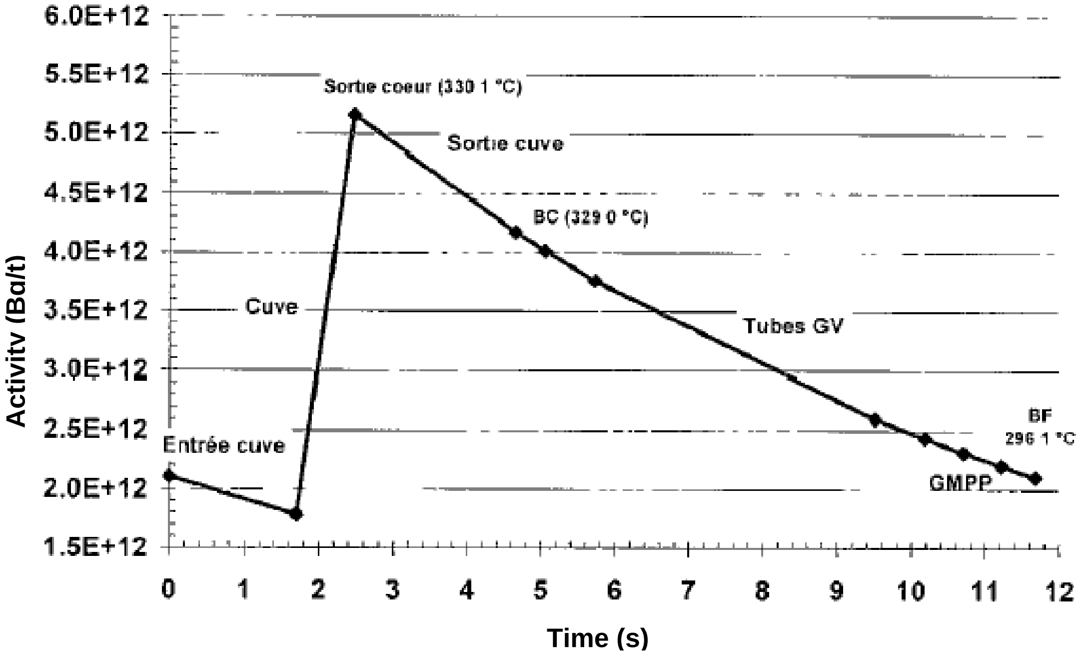
Specific Nuclide Concentrations in the Primary System (RCP [RCS] 1, 2, 3, 4) – Shutdown Transient [Ref-1]

NUCLIDE	SPECIFIC ACTIVITY (MBq/t)		
	Realistic	Biological protection design	Effluent treatment system design
Mn-54	2000	3700	1400*
Co-58	160000	250000	10000*
Fe-59	9700	37000	360*
Co-60	3300	5900	580*
Cr-51	18000	36000	9500*
Ag-110m	7200	16000	1700*
Sb-122	7100	10000	1100*
Sb-124	3000	3700	560*
SB-125	510	1000	100*
Ni-63	3100	3100	3100
Kr-85m	460	13000	31000
Kr-85	73	1200	4300
Kr-87	830	23000	30000
Kr-88	1200	32000	45000
Xe-131m	53	830	3100
Xe-133m	260	3900	23000
Xe-133	9500	150000	550000
Xe-135	1900	25000	130000
Xe-138	2500	41000	72000
I-131	2300	37000	110000
I-132	2200	34000	82000
I-133	2400	37000	210000
I-134	1500	24000	30000
I-135	1400	23000	140000
Cs-134	960	7700	34000
Cs-136	120	360	37000
Cs-137	800	6400	25000
Cs-138	2500	41000	100000
Sr-89	30	490	3000
Sr-90	0.19	3	19
N-16	*see Sub-Chapter 6.1 - Figure 1*		
N-17	*See Sub-Chapter 6.1 - Figure 2*		
Ar-41	460	1000	3000
H-3	37000	37000	37000
C-14	6	13	13

*: for corrosion products, two effluent treatment system design source terms were calculated: one at the time of reduction of load and one at the oxygenation peak [Ref-1]. However in Table 2, only the source term calculated at the time of reduction of load is presented, because the “oxygenation peak” source term has not been used [Ref-2].

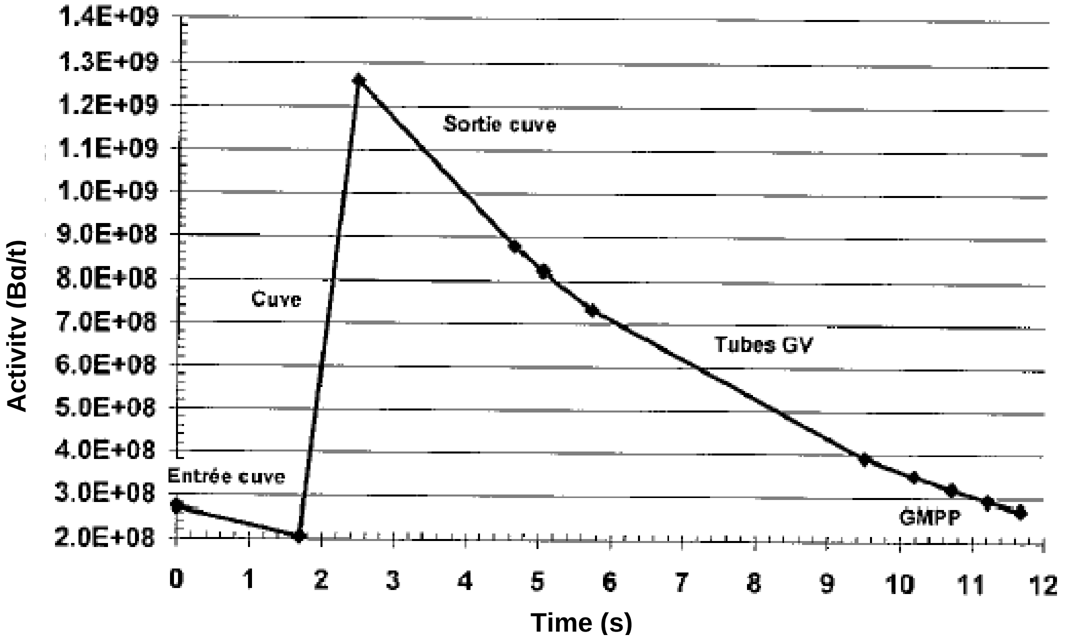
SUB-CHAPTER 6.1 - FIGURE 1

Nitrogen-16 Content in the Primary Loops [Ref-1]



SUB-CHAPTER 6.1 - FIGURE 2

Nitrogen-17 Content in the Primary Loops [Ref-1]



SUB-CHAPTER 6.1 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

[Ref-1] Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs. The Environment Agency. January 2007. (E)

1. DEFINITIONS OF THE SOURCE TERMS USED

[Ref-1] Primary Source Term of the EPR Reactor. ENTERP090062 Revision A. EDF. March 2009. (E)

ENTERP090062 Revision A is the English translation of ENTERP070147 Revision A.

[Ref-2] Use of source term in the different GDA areas. ECEIG101686 Revision B. EDF. November 2010. (E)

2. ISOTOPE INVENTORY

2.1. NITROGEN-16

[Ref-1] Primary Source Term of the EPR Reactor. ENTERP090062 Revision A. EDF. March 2009. (E)

ENTERP090062 Revision A is the English translation of ENTERP070147 Revision A.

2.3. TRITIUM

[Ref-1] Primary Source Term of the EPR Reactor. ENTERP090062 Revision A. EDF. March 2009. (E)

ENTERP090062 Revision A is the English translation of ENTERP070147 Revision A.

2.5. CARBON-14

[Ref-1] Zinc Injection claims, arguments and evidences: overall balance for UK-EPR. ECEF110139 Revision A. EDF. March 2011. (E)

[Ref-2] Primary Source Term of the EPR Reactor. ENTERP090062 Revision A. EDF. March 2009. (E)

ENTERP090062 Revision A is the English translation of ENTERP070147 Revision A.

2.6. CORROSION PRODUCTS

[Ref-1] Analysis of UK EPR™ Source Term: Identification, Quantification and Characterisation. ECEF110448 Revision A. EDF. July 2011. (E)

[Ref-2] Corrosion product characterization under PWR/EPR primary coolant conditions: thermodynamic assessments and power plant feedback. ECEF111022 Revision A. EDF. July 2011. (E)

2.7. FISSION PRODUCTS

[Ref-1] Analysis of UK EPR™ Source Term: Identification, Quantification and Characterisation. ECEF110448 Revision A. EDF. July 2011. (E)

[Ref-2] Corrosion product characterization under PWR/EPR primary coolant conditions: thermodynamic assessments and power plant feedback. ECEF111022 Revision A. EDF. July 2011. (E)

SUB-CHAPTER 6.1 - TABLES 1 AND 2

[Ref-1] Primary Source Term of the EPR Reactor. ENTERP090062 Revision A. EDF. March 2009. (E)

ENTERP090062 Revision A is the English translation of ENTERP070147 Revision A.

[Ref-2] Use of source term in the different GDA areas. ECEIG101686 Revision B. EDF. November 2010. (E)

SUB-CHAPTER 6.1 - FIGURES 1 AND 2

[Ref-1] Primary Source Term of the EPR Reactor. ENTERP090062 Revision A. EDF. March 2009. (E)

ENTERP090062 Revision A is the English translation of ENTERP070147 Revision A.