MDEP

Technical Report

TR-EPRWG-03

EPR Working Group

Technical Report on the definition of primary coolant <u>source terms</u> used in the different EPR designs for shielding, radiation zoning, DBA consequences…

Participation

Regulators involved in the MDEP working group discussions:	NNSA (China), STUK (Finland), ASN (France), AERB (India), ONR (U.K.), and US NRC (U.S.A.)
Regulators which support the present report	NNSA (China), STUK (Finland), ASN (France), AERB (India), ONR (U.K.), US NRC (U.S.A.) and SSM (Sweden)
Compatible with existing IAEA related documents:	Yes

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Summary

1. INTRODUCTION

In May 2010, during the 6th EPR Working Group (EPRWG) meeting [1], the co-chair of the group (ASN) expressed the interest of the EPR's Family to understand reasons behind differences in practices and regulations in application in different countries on several issues, including Reactor coolant system (RCS) source term.

It was suggested to have further discussions between regulators to better understand differences on RCS source terms as well as impact on shielding and operation.

In September 2010 [2], it was agreed that the EPRWG Radiation Protection Experts would conduct a survey of all the EPRWG members to document how the different designs determine the source term for shielding, radiation zoning and room accessibility.

This document aims at:

- understanding how the reactor coolant system source term is defined for the different EPR designs,
- identifying differences and explain their origin,
- sharing good practices to limit primary source term during the design stage.

This report has been established on the basis of answers given by EPRWG members to a questionnaire elaborated by IRSN (France). The following countries have participated: China, France, Finland, U.K. and U.S.

2. GENERALITIES ON SOURCE TERM

Source term is the quantity of each kind of radioactive material and the physical and chemical speciation that is assumed to be produced and released from the core. In general, source term includes all radionuclides coming from activation or fission products, including those inside the fuel pellets. However, in this document, source term is used for radionuclides coming from activation, corrosion or fission products contained in the reactor coolant system, not taking into account fission products contained inside the fuel.

Source terms discussed in this document are used to design the installation and to assess design basis accident consequences for accident that do not lead to cladding failures.

The RCS source term is composed of:

- fission products, issued from small defects in fuel cladding during plant operation,
- activation products of the main coolant, as tritium or nitrogen 16,

- corrosion products released by internal structures of the primary circuit and which are activated when going through the core active zone, as cobalt-60.

The source term depends on the reactor state which could be a normal power state, a transient power state including the chemical conditioning of the primary coolant, or an accidental state. It corresponds to the sum of activities in the reactor coolant or deposits inside the main coolant pipes.

During the operation at full power, activity is mainly due to N-16 and N-17 radionuclides, which have very short radioactive periods. Therefore, N-16 and N-17 activity disappears when the neutron flux is stopped, being nil when aerating the primary coolant prior to the main coolant system opening. During reactor shutdown states, corrosion products which have been activated in the core are responsible for almost 90% of integrated doses during workers interventions. Then the choice of equipment materials plays a key role in the source term composition regarding corrosion products.

For design basis accident (DBA), the source term in steady-state and transient conditions has to be assessed to calculate the radiological consequences of accidents with no core degradation in the frame of accident studies.

3. ACTIVITIES SPECTRUMS CONSIDERED FOR EPR DESIGN

Spectrums used in the different countries having participated to the survey are successively described. Then some conclusions are drawn up.

Finland

Initially two source terms were proposed by TVO/CFS in the Olkiluoto 3 Preliminary safety analysis report (PSAR) and supporting documents:

- the so-called DESIGN source term covering 95 % of the operation time covering value of French and German NPPs, used for the design of biological protection (walls, shields),
- the so-called TYPICAL source term representing typical operation conditions especially maintenance phases (expected average values in “Best-Estimate” conditions).

The definition of these source terms is different for the purposes of occupational radiation protection, design and radioactive release analyses.

According to STUK’s requirement also so-called « TechSpec » radionuclide concentrations of the primary coolant were defined for accident analyses. The « TechSpec » concentration values were defined to be 5 x DESIGN for fission products, and equal to DESIGN for activation and corrosion products, and they will be presented in the Safety Technical Specifications of Olkiluoto 3 as the maximum allowed concentrations.

France

Initially, two activities spectrums were also proposed by EDF in the Flamanville 3 preliminary safety report:

- the first one (so-called TYPICAL) used for studying the classification of nuclear pressurized equipment and for the provisional dose calculations;
- the second one (so-called DESIGN) used for designing biological protections and for assessing radiological consequences of accidents without core melt.

In the frame of the safety analysis of the PSAR in 2006, IRSN has estimated that the “DESIGN” source term was not sufficient to design and size radioactive effluents treatment systems. Therefore, EDF has revised primary coolant radionuclides inventory as a function of EPR operating phases. The revised inventory could then be used as a common reference for all EPR design studies and guarantee the consistency between the PSAR different chapters. A third source term has been created by EDF to perform design studies for effluents treatment systems and the evaluation of radiological consequences for design basis accident without core damage.

Then, there is now three source terms considered for the FA3 EPR design, respectively for:

- Designing nuclear pressurized equipment and provisional evaluation of workers doses (realistic source term), covering 95% of the time covering value of French N4 1450 MWe NPPs,
- Designing and sizing biological protections (shielding), covering 100% of the time covering value of French N4 1450 MWe NPPs,
- Designing and sizing radioactive effluents treatment systems and evaluating radiological consequences for accident without core damage[1], covering 100% of the time covering value of French N4 1450 and 1300 MWe NPPs (except for corrosion products).

UK

As for France, 3 activities spectrums have been defined:

- <u>Realistic source term</u> - The realistic source term is representative of the average specific activities most likely to be seen during normal operating conditions. This source term was defined in order to define the French ESPN (nuclear pressure equipment) classifications and

[1] This source term is used to assess the radiological consequences of accidents that do not lead to cladding failures except those pre-existing, i.e. which occurred during normal operation (for example: SGTR).

to perform initial worker dose assessments. This source term encompasses the average values measured on the French N4 1450 MWe series,

- Biological shielding design source term - The biological shielding design source term is more conservative than the realistic source term. It corresponds to specific activity values covering all spectrometry measurements obtained on the French N4 1450 MWe series. It is used to design and size the rooms, systems and shielding,
- Effluent treatment system design source term - This source term is used in the design of treatment systems for effluents (filtration, demineralization and evaporation) and radioactive wastes as well as for radiological impact assessment in accidents (DBA and PSA). It covers all the spectrometry measurements obtained in the French 1300 MWe and N4 1450 MWe series, which includes fuel failures.

US

There are four source terms identified in the Final Safety Analysis report for the US EPR. They are:

- Design basis used for the design of the radioactive waste management systems, assuming 1% failed fuel per Standard Review Plan (NUREG-0800) §11.2 §11.3 [6] with objective of demonstrating "defence in depth" when compared to ANSI/ANS18.1 [8].
- Normal operations used to estimate annual effluent concentrations, ANSI/ANS 18.1 as used in LADTAPII and GASPARII codes per SRP §11.2 §11.3, respectively.
- Design basis used for normal operation equipment qualification and shielding calculations to protect plant workers, per Regulatory Guide 8.8 [7].
- The initial coolant conditions for design basis accident consequences analysis. Same as shielding applies 0,25% failed fuel per RG 8.8.

China

Initially, 3 activities spectrums were considered for Taïshan EPR:

- REALISTIC values of source term used for workers' dose assessment,
- DESIGN values of source term used for the design of nuclear pressurized equipment and the evaluation of biological shielding,
- TECH-SPEC values of source term used for radiological assessment following accidents, verification of environmental safety under extreme conditions beyond the design basis, and shielding design of waste treatment systems.

Following to customer request, two new sets of values are currently applicable:

- REALISTIC: source term based on realistic activity levels in the reactor coolant during normal operation of recent French and German NPPs. These are averaged values expected in “best-estimate” operating conditions. This source term is mainly used for determining the worker doses during maintenance tasks (dose assessment) and to justify the collective dose target.

- DESIGN 0.25: source term of the fission products (FP) calculated with 0.25 % fuel rod defects for the design source term. This source term corresponds to the covering values measured on French and German NPPs for activation products, activated corrosion products and tritium activities.

Conclusion

The following scheme summarizes the use of the different source terms for the different EPR:

Finland	France, UK	US	China
Design of process systems (including nuclear pressurized equipment) and radiation shielding	Design of nuclear pressurized equipment and provisional evaluation of workers doses	Design of the radioactive waste management systems	Workers’ dose assessment during maintenance tasks and justification of the collective dose target
Evaluation of occupational radiation doses	Design of biological protections (shielding)	Shielding calculations to protect plant workers and Equipment qualification	Design of nuclear systems, evaluation of biological shielding
Evaluation of radiological consequences of accidents	Design and sizing effluents treatment systems and evaluation of radiological consequences for accident without core damage	Normal operations used to estimate annual effluent concentration	Radiological assessment following accidents, design of waste treatment systems
		Initial coolant conditions for design basis accident consequence analysis	

It can be noted that in China, France and UK, objectives fixed for the different source terms are similar, even if the design of nuclear systems is based on a more conservative source term in China. In Finland, the situation is quite similar as well, the design of nuclear pressurized equipment is also based on DESIGN source term but methodology is different. In France and UK, the use of DESIGN source term to design nuclear pressure equipment is due to French nuclear pressure equipment classification [3]. In China, DESIGN 0.25 source term can be used to design the systems, being in compliance with construction permit.

Source term is used also in US to design the reactor coolant system. But it is not used to evaluate the occupational radiation doses. Moreover, unlike the other countries, a source term is dedicated to estimate annual effluent concentration.

4. RATIONALES FOR SOURCE TERM DEFINITION

Finland	Operating feedback from reference NPPs
France	Realistic source term coverage: 95% of equivalent activities of 1450 MWe NPPs Biological shielding design source term coverage: 100% of equivalent activities of 1450 MWe NPPs Effluent treatment system design source term: 100% of equivalent activities on 1300 and 1450 MWe NPPs (to cover possible cladding failures during normal operation) For the radiological consequences assessment of DBA: Operating feedback (1300 and 1450 MWe NPP)[2] - A specific methodology is used to build the source terms. For most the fission products, activities are normalized to the radiochemical specifications (equivalent iode-131 to 20 GBq/t in steady-state operation and 150 GBq/t during power transients)
U.K.	Majority of source term data is derived from operational feedback. Limited data is also derived from calculations and estimations. Realistic source term: feedback from mean values measured on the French 1450 MWe plants, covering 95% of values. Biological shielding design source term: feedback from maximum values measured on the French 1450 MWe plants. Effluent treatment system design source term: fission products - activity values are normalized with the radiochemical specifications of existing plants which cover the spectrometry measurements obtained on French 1300 and 1450 MWe plants. This includes cycles with fuel failures. Corrosion products - maximum values measured at the French 1450 MWe plants. Deposited activity by comparison to 1300 and 1450 MWe plants.

[2] for α and β emitters, feedback from specific measuring campaign are used to define activities

U.S.	Design basis used for the design of the radioactive waste management systems: based on 1% failed fuel and calculations for the core inventory from the ORIGEN code. Activation products and tritium concentrations are derived from ANSI 18.1. Design basis secondary coolant concentrations based on the technical specifications limit primary to secondary leak rate of 600 gpd (EPR specific) for four steam generators. Normal operations used to estimate annual effluent concentrations: based on ANSI 18.1 and calculations using the GALE code. The GALE code is for evaluating the LWS and GWS performance. The results (output) of GALE code is used as input in LADTAPII and GASPARII for dose assessments due to the effluents. Design basis used for normal operation equipment qualification, quality group classification of rad waste systems and components, and shielding calculations to protect plant workers: based on 0.25% failed fuel and calculations from the ORIGEN code. Activation products and tritium concentrations are derived from ANSI 18.1. Design Basis Accident Analysis: based on regulatory guide 1.183 also known as the Alternative Source Term (AST). The initial RCS concentration may be based on the 1% failed fuel and calculations from the ORIGEN Code. Activation products and tritium concentrations are derived from ANSI 18.1. The AST LOCA assumes an extensive in-vessel core damage releasing 100% of the NB and 40% halogens along with other fission products and actinides. The initial (pre-accident) RCS concentration is then a small contributor to dose.
China	Operating feedback (French 1300 MWe NPP and German Konvoi plants) REALISTIC source term: averaged values expected in ‘Best-Estimate’ operating conditions DESIGN 0.25: Source term of the fission products (FP) calculated with 0.25 % fuel rod defects for the design source term. Activation products, activated corrosion products and tritium activities: all covering values measured for French and German plants. For tracking purpose, DESIGN values: correspond to maximum of time covering values measured on French and German plants. TECH SPEC values coverage: several times higher than DESIGN values.

Analysis

For China, Finland, France and UK, source term is essentially based on operating feedback. The French 1300 and 1450 MWe NPP operating feedback is used for France and UK. The source term defined for Taïshan EPR takes as well into account the feedback from Konvoï reactors. It would be interesting to compare the activities obtained in both cases.

For the US EPR, only the source term used for estimating the effluent activities is based on operating feedback, using a dedicated methodology, except for the demonstration of defence in depth of the LWS and GWS that assume even with 1% failed fuel one can meet the 10CFR 20 Appendix B concentration limits [9]. The determination of the « Design Basis for Shielding and Design Basis Accident Analysis » initial condition for the source term is based on a lump-sum approach, corresponding to 0.25% of cladding failure, except for radioiodines and bromines and noble gases which are assumed to be on the TechSpec dose equivalent (DE) I-131 and Xe-133 limits. The accident initiated source term is the AST. This approach is used as well in China.

N.B The 0,25% fuel failure is 2x the NUREG-0017 Revision 0 (LADTAP) [4] operating experience (REX) which represented an equivalent fuel failure fraction of 0,125%.

As a conclusion, it can be noticed that there is mainly two approaches:

- The first one is based on operating feedback,
- The second one is based on a lump-sum approach, corresponding to a given percentage of cladding failure (0.25% or 1% depending on the source term use) except for some radionuclides.

For China, Finland, France and UK, steady-state operation, shutdown transients and shutdown conditions are systematically considered for the definition of activities spectrums. For US, only the steady state operation is taken into account.

5. ACTIVATED CORROSION PRODUCTS

The list of corrosion products taken into account in the reactor coolant system source terms is described in the table below:

Finland	Mn-54, Co-58, Fe-59, Co-60
France	Mn-54, Co-58, Fe-59, Co-60, Cr-51, Ni-63, Ag-110m, Sb-122, Sb-124, Sb-125 For the radiological consequences assessment of DBA: Mn-54, Co-58, Fe-59, Co-60, Cr-51, Ni-63, Ag-110m, Sb-122, Sb-125
U.K.	Mn-54, Co-58, Fe-59, Co-60, Cr-51, Ni-63, Ag-110m, Sb-122, Sb-124, Sb-125

U.S.	Na-24, Cr-51, Mn-54, Fe-55, Co-58, Fe-59, Co-60, Zn-65, Ag-110m, W-187
China	Cr-51, Mn-54, Fe-59, Co-58, Co-60, Ni-63, Ag-110m, Sb-122, Sb-124, Sb-125

It can be noticed that the list of corrosion products considered in China, France and UK is similar.

In Finland, the list is limited to the most important products. Co-60 and Co-58 radionuclides are the main contributors to the dose rate in shutdown conditions:

- Co-60 comes from Co-59 activation; Co-59 is mainly found in hard alloy with cobalt (stellite and Haynes 25) and, in a lesser extent, in nickel-based alloy and stainless steel where its concentration is limited,
- Co-58 comes from the activation of Ni-58; Ni-58 is the main component of 690 alloy and, in a lesser extent, in stainless steel.

Efforts have been made by vendors to limit the use of Cobalt based alloys in EPR components in order to reduce the production of Co-58 and Co-60. Main design innovations are described below. Therefore, the contribution of other corrosion products has been correlatively increased.

Design innovations considered in EPR to limit Co-60 and Co-58 radionuclides

Finland	▪ Reduction in surface areas of cobalt-based alloys ▪ Reduction of specified cobalt impurities in nickel alloys or stainless steel ▪ Design of the CVCS (primary coolant purification) ▪ Chemical conditions of the primary circuit ▪ Chemical passivation of surfaces
France	▪ Reduction in surface areas of cobalt-based alloys ▪ Electro-polishment of SG water boxes ▪ Design of the CVCS (purification function) ▪ Procedure for the RCS passivation ▪ Chemical conditioning (zinc injection, use of boric acid enriched boron 10, pH control of the reactor coolant) ▪ Procedure to reach a cold shutdown state and start-up of the reactor

U.K.	▪ Reduction in high cobalt alloy inventory ALARP (< $2m^2$, excluding CRDMs) ▪ Minimising residual cobalt impurities, especially in SG tubes ▪ Exclusion as far as possible of silver and antimony in components in contact with the primary coolant ▪ Optimised manufacturing process for SG tubes ▪ Hot Functional Testing and conditioning of primary circuit prior to fuel loading ▪ Primary coolant chemistry control (normal operations and start-up and shutdown), including zinc injection
U.S.	Same as in France

Steam generator electro-polishment

Electro-polishment is a surface treatment which consists in eliminating a part of the thickness of the material by an electrical and chemical process in order to optimize the state of the surface and to reduce the release of materials and the deposition of corrosion products. In France, it is considered that the gain on the dose rate may be about 40%. The table below indicates if this option has been or not considered in the different EPR.

Finland	France	U.K.	U.S.	China
No	Yes	No This is claimed on the basis of lower dose rates and improved maintenance practices	Will be determined by each licensee	Yes

6. MAIN COOLANT ACTIVATION PRODUCTS

Hereafter are listed the products resulting from reactor coolant activation and considered in the definition of source terms in the different countries.

Finland	N-16, N-17, H-3, Ar-41, C-14 For accident analyses: Ar-41
France	N-16, N-17, H-3, Ar-41, C-14 For the radiological consequences assessment of DBA: Ar-41
U.K.	N-16, N-17, H-3, Ar-41, C-14
U.S.	N-16, N-17, H-3, Ar-41, C-14
China	N-16, N-17, H-3, Ar-41, C-14

The list of activation products considered in EPR source terms is quite the same for all countries.

Volume activity of H-3 in the main coolant depends on the production inside the primary circuit but also on the global strategy of management of H-3 in the installation. This strategy relies mainly on the politic of management of tritiate wastes.

In France, all H-3 produced during a year is released into the environment, in the respect of discharge decree requirements: the possibility to recycle primary H-3 effluents in the primary circuit is just used in case of difficulties on discharge conditions in low waters or flood period.

7. FISSION PRODUCTS

7.1 LIST OF FISSION PRODUCTS TAKEN INTO ACCOUNT

In this section are listed the fission products considered for the definition of source terms, i.e. all fission products considered as a whole independently from the different type of source terms.

Finland	For accident analyses: Kr-85m, Kr-85, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135, Xe-138, Sr-89, Sr-90, I-131, I-132, I-133, I-134, I-135, Cs-134,Cs-137
France	For the radiological consequences assessment of DBA: Kr-85, Kr-85m, Kr-87, Kr-88, Xe-131m, Xe-133, Xe-133m, Xe-135, Xe-135m, Xe-138, I-131, I-132, I-133, I-134, I-135, Te-132, Cs-134, Cs-136, Cs-137, Cs-138, Rb-88, Sr-89, Sr-90, Ce-144, Ru-103, Ru-106, Np-239 and α emitters[3]
U.K.	Kr, Xe, Sr, I and Cs isotopes – Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135, Xe-138, Sr-89, Sr-90, I-131, I-132, I-133, I-134, I-135, Cs-134, Cs-136, Cs-137, Cs-138
U.S.	Kr-83m, Kr-85m, Kr-85, Kr-87, Kr-88, Kr-89, Xe-131m, Xe-133, Xe-135m, Xe-135, Xe-137, Xe-138, , I-131, I-132, I-133, I-134, I-135, , Cs-134, Cs-136, Cs-137, Cs-138, Sr-89, Sr-90, Sr-91, Sr-92,
China	Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135, Xe-138, Sr-89, Sr-90, I-131, I-132, I-133, I-134, I-135, Cs-134, Cs-136, Cs-137, Cs-138, Sr-91, Sr-92, Y-90, Y-91, Zr-95, Nb-95, Mo-99, Tc-99m, Te-131m, Te-131, Te-132, Te-134, Ba-140, La-140, Ce-141, Ce-143, Ce-144, Pr-143, Pr-144, Ru-103, Ru-106

[3] Xe-135m, Te-132, Ce-144, Ru-103, Ru-106, Np-239 and α emitters are just for DBA.

7.2 RATIONALES FOR FISSION PRODUCTS SOURCE TERM ASSESSMENT

Finland	Operating feedback from the reference NPPs
France	For the radiological consequences assessment of DBA: operating feedback (900, 1300 and N4 NPP)[4] - Specific methodology is used to build the source terms. For most of the fission products, activities are normalized to the chemical and radiological specifications (equivalent iode-131 to 20 GBq/t in steady-state operation and 150 GBq/t during power transients)
U.S.	Operating limits (technical specifications), calculations, and industry standards
China	At first, the values for predominant nuclides have been determined according to experience. The activity concentrations of the other fission products were deduced from the available spectrum using normalized design values. Then, calculations using 0.25% of fuel rods defect are performed to evaluate the fission products spectrum.

[4] As already mentioned, for α and β emitters, feedback from specific measuring campaign are used to define activities.

When fission products source term is coming from operating feedback, NPP considered are given below:

	Finland	France	U.K.	U.S.	China
NPP considered	Some reference NPPs – no detailed data available	1300 and 1450 MWe NPP	Initially French 900 MWe plants and calculations, later: French 1300 and 1450 MWe plant	Industry Standard ANS/ANSI 18.1-1999 based on NPP feedback	At first, French and German plants
Fuel management considered in statistical treatment?	?	Yes	Yes	Yes (MOX, Enrichment, Burn-up sensitivities are performed)	
Are measurement uncertainties taken into account?	?	No	No	Power level uncertainty is taken into account as well as fuel manufacturing tolerances.	

7.3 NORMALIZATION OF FISSION PRODUCTS SOURCE TERM TO THE RADIOCHEMICAL SPECIFICATIONS

Finland	France	U.K.	U.S.	China
Yes, the fission product source term (TechSpec concentrations) has been normalized to the radiochemical specifications (I-131 concentration 8 GBq/t).	Yes, for the radiological consequences assessment of DBA. For most of the fission products, activities are normalized to the chemical and radiological specifications (equivalent iode-131 to 20 GBq/t in steady-state operation and 150 GBq/t during power transients)	Yes, for Effluent treatment system design source term. Normalised to DE I-131 at 20 GBq/t in normal operation and 150 GBq/t during transients.	Yes, for DBA analysis. Normalized to Technical Specification 1 μCi/gm DE I-131 and 210 μCi/gm DE XE-133 for steady-state operation DBA dose analysis assumes second case where initial primary coolant concentration is at 60 μCi/gm DE I-131 for transients	Yes

7.4 SOURCE TERMS ASSESSMENT FOR POWER TRANSIENTS

The activity during power transients can be assessed by using peak factor or operating feedback. The following table stipulates the way power transient is modelled for the different EPR:

Finland	France	U.K.	U.S.	China
Peak factor	Experience feedback	See below	Based on specific guidance given in NRC regulatory guides	Operational feedback

The table below gives some information on the duration of the activity peak after transient considered for the radiological consequences assessment of DBA (height of the peak, how the peak is assumed to decrease).

Finland	U.K.	France	U.S.	China
Exponential increase with a doubling time of 10 minutes up to 30 times the initial concentration for relevant nuclides (I-131, Cs-134, Cs-137), then decrease according to the characteristics of the purification system.	See below Duration: 30 or 90 minutes depending upon analysis.	Duration:1h30	Accident specific - Two iodine spiking cases. See below.	

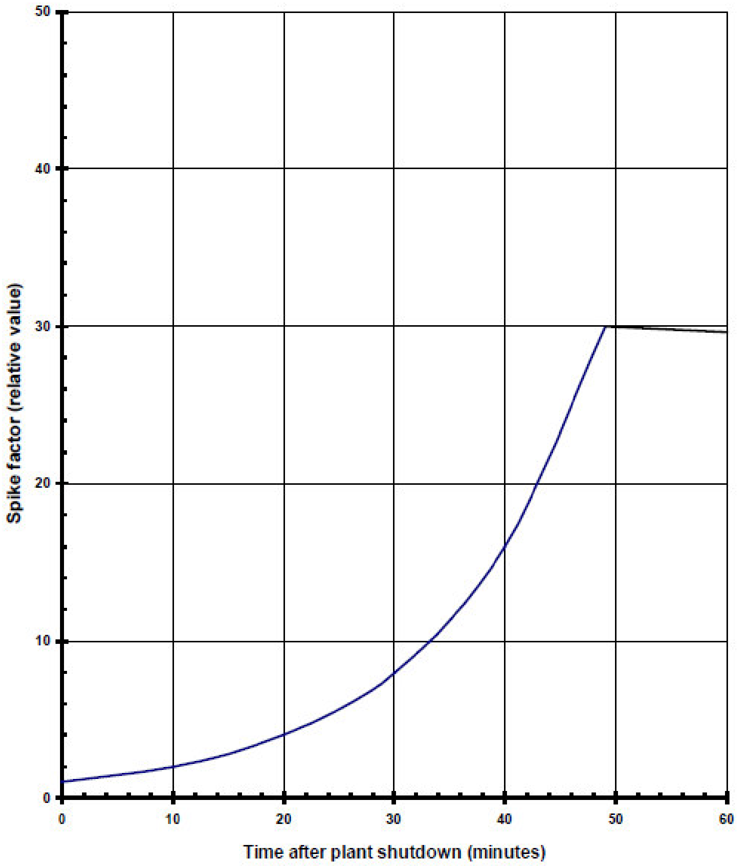
Finland

Table 12.2-1: Nuclide Specific Concentrations in RCP1, 2, 3 / JEA, JEB, JEC – Shutdown Spiking Factors

NUCLIDE	SPECIFIC ACTIVITY (Bq/Mg)			SHUTDOWN
	DESIGN	TYPICAL	TechSpec	SPIKING RATIO
Mn-54	4.0E+06	2.0E+06	4.0E+06	300
Co-58	1.6E+07	8.0E+06	1.6E+07	10000
Fe-59	1.0E+06	5.0E+05	1.0E+06	300
Co-60	1.0E+06	5.0E+05	1.0E+06	500
Ar-41	1.0E+09	3.0E+08	1.0E+09	1.0
Kr-85m	5.5E+09	2.0E+08	2.8E+10	2.3
Kr-85	5.2E+08	1.9E+07	2.6E+09	1.0
Kr-87	1.0E+10	3.6E+08	5.0E+10	2.3
Kr-88	1.4E+10	5.0E+08	7.0E+10	2.3
Xe-133m	1.7E+09	1.1E+08	8.5E+09	2.3
Xe-133	8.0E+10	5.0E+09	4.0E+11	1.9
Xe-135	1.8E+10	1.1E+09	9.0E+10	1.4
Xe-138	1.4E+10	8.5E+08	7.0E+10	2.9
Sr-89	4.9E+06	3.0E+05	2.5E+07	1.0
Sr-90	3.0E+04	1.9E+03	1.5E+05	1.0
I-131	1.6E+09	1.0E+08	8.0E+09	23
I-132	2.8E+09	1.8E+08	1.4E+10	12
I-133	4.9E+09	3.1E+08	2.5E+10	7.6
I-134	1.7E+09	1.1E+08	8.5E+09	14
I-135	3.3E+09	2.0E+08	1.7E+10	7.1
Cs-134	3.2E+08	4.0E+07	1.6E+09	24
Cs-137	3.2E+08	4.0E+07	1.6E+09	20
Cs-138	1.4E+10	8.5E+08	7.0E+10	2.9
N-16	*See Figure 12.2-1*		-	-
H-3	3.7E+10	-		-

UK

From "UK EPR PCSR Chapter 14.6". Assumed relative increase of specific activity of I-131, Cs-134 and Cs-137 in the primary coolant after plant shutdown.



Radiological protection

From "Primary Source Term of the EPR Reactor. ENTERP090062 Revision A. EDF. March 2009."

Nuclide	Specific Activity / MBq t^{-1}				Notes
	Biological shielding design source term		Effluent treatment system design source term		
	Normal Operations	Transients	Normal Operations	Transients	
H-3	3.70E+04	3.70E+04	3.70E+04	3.70E+04	Bounding assumption
C-14	1.30E+01	1.30E+01	1.30E+01	1.30E+01	Calculated for UK EPR based on French 1300 MWe feedback
N-16	Dependant on position around RCS; maximum 5.2E+12				

N-17	Dependant on position around RCS; maximum 1.26E+09				
Ar-41	1.00E+03	1.00E+03	3.00E+03	3.00E+03[5]	
Cr-51	6.00E+02	3.60E+04	6.00E+02	3.60E+04	
Mn-54	2.20E+02	3.70E+03	2.20E+02	3.70E+03	
Co-58	3.90E+02	2.50E+05	3.90E+02	2.50E+05	
Fe-59	8.10E+01	3.70E+04	8.10E+01	3.70E+04	
Co-60	1.70E+02	5.90E+03	1.70E+02	5.90E+03	
Ni-63	1.50E+01	3.10E+03	1.50E+01	3.10E+03	Based on single measurement campaign at a French 1300 MWe plant
Kr-85m	5.50E+03	1.30E+04	1.50E+04	3.10E+04	
Kr-85	6.20E+02	1.20E+03	2.40E+03	4.30E+03	Calculated based on Xe-133 activity
Kr-87	1.00E+04	2.30E+04	2.30E+04	3.00E+04	
Kr-88	1.40E+04	3.20E+04	3.50E+04	4.50E+04	
Sr-89	4.90E+00	4.90E+02	3.00E+01	3.00E+03	Calculated for UK EPR based on PROFIP 5.1 code
Sr-90	3.00E-02	3.00E+00	1.90E-01	1.90E+01	Calculated for UK EPR based on PROFIP 5.1 code
Ag-110m	2.70E+02	1.60E+04	2.70E+02	1.60E+04	
Sb-122	1.10E+02	1.00E+04	1.10E+02	1.00E+04	
Sb-124	1.20E+02	3.70E+03	1.20E+02	3.70E+03	
Sb-125	9.80E+01	1.00E+03	9.80E+01	1.00E+03	
Xe-131m	4.40E+02	8.30E+02	1.70E+03	3.10E+03	Calculated based on Xe-133 activity
I-131	1.60E+03	3.70E+04	1.50E+04	1.10E+05	

[5] In UK PCSR 11.1 - For corrosion products (Effluent treatment system design only, Ar-41, Cr-51, Mn-54, Co-58, Fe-59, Co-60, Ag-110m, Sb-122, Sb-124), the source term was calculated at the time of reduction of load rather than at the oxygenation peak

I-132	2.80E+03	3.40E+04	1.80E+04	8.20E+04	
Xe-133m	1.70E+03	3.90E+03	1.10E+04	2.30E+04	
I-133	4.90E+03	3.70E+04	2.40E+04	2.10E+05	
Xe-133	8.00E+04	1.50E+05	3.10E+05	5.50E+05	
Cs-134	3.20E+02	7.70E+03	4.50E+03	3.40E+04	
I-134	1.80E+03	2.40E+04	7.70E+03	3.00E+04	
I-135	3.30E+03	2.30E+04	1.60E+04	1.40E+05	
Xe-135	1.80E+04	2.50E+04	9.20E+04	1.30E+05	
Cs-136	3.30E+01	3.60E+02	2.10E+03	3.70E+04	
Cs-137	3.20E+02	6.40E+03	3.30E+03	2.50E+04	
Cs-138	1.40E+04	4.10E+04	1.00E+05	1.00E+05	
Xe-138	1.40E+04	4.10E+04	7.20E+04	7.20E+04	

The “Realistic source term” was used to estimate occupational doses, whereas the “Biological shielding design source term” above was used as a design parameter for buildings, systems and shielding provisions in the UK EPR.

Accident Analysis

Site specific calculations for design basis radiological consequences are out of scope of General design assessment (GDA). The intention of EDF and AREVA has always been to provide site specific calculations at a later date as part of nuclear site licensing.

The source term is used in the assessment of DBA and PSA events. The radiological consequences analysis presented in Chapter 14.6 of the PCSR makes assumptions established during the basic design phase of the EPR project and are partly based on German regulations. The calculations are based on the Flamanville 3 EPR Preliminary Safety Report (PSR) and are based on the original EPR source term which considered only two source terms, “Design” and “Typical”. These correspond to the “Realistic source term” and the “Biological shielding design source term” as currently defined for UK EPR, prior to the update. “Design” values are used for accident analyses presented in the PCSR. These values are tabulated below;

Nuclide	Specific Activity / MBq t^{-1}		
	Design	Typical	Spiking Factor
H-3	3.70E+04	3.70E+04	1
N-16	5.70E+06	4.30E+06	0
Ar-41	1.00E+03	3.00E+02	1
Mn-54	4.00E+00	2.00E+00	300
Co-58	1.60E+01	8.00E+00	1000
Fe-59	1.00E+00	5.00E-01	300
Co-60	1.00E+00	5.00E-01	500
Kr-85m	5.50E+03	2.00E+02	2.3
Kr-85	5.20E+02	1.90E+01	1
Kr-87	1.00E+04	3.60E+02	2.3
Kr-88	1.40E+04	5.00E+02	2.3
Sr-89	4.90E+00	3.00E-01	1
Sr-90	3.00E-02	1.90E-03	1
I-131	1.60E+03	1.00E+02	30
I-132	2.80E+03	1.80E+02	12
Xe-133m	1.70E+03	1.10E+02	2.3
I-133	4.90E+03	3.10E+02	7.6
Xe-133	8.00E+04	5.00E+03	1.9
Cs-134	3.20E+02	4.00E+01	30
I-134	1.70E+03	1.10E+02	14
I-135	3.30E+03	2.00E+02	7.1
Xe-135	1.80E+04	1.10E+03	1.4
Cs-137	3.20E+02	4.00E+01	30
Cs-138	1.40E+04	8.50E+02	2.9
Xe-138	1.40E+04	8.50E+02	2.9

To provide additional information to the French regulator, the PCSR also includes a limited number of calculations utilising assumptions consistent with those used by EDF in the assessment of the radiological consequences in their French fleet. These calculations include a factor which increases the source term considered to levels that are the same as current EDF technical specification levels in operating plants, namely 20 GBq t^{-1} DE I-131 during normal operations and 150 GBq t^{-1} DE I-131 following a transient. This method results in a multiplication for all fission product source terms of 6 during normal operations and 2.5 during transients. This source tem is then closer to the "Effluent treatment system design source term" defined above.

With respect to the dose calculation all nuclides are selected except N-16 (very short half-life) and ^{3}H (minor radiological relevance).

U.S.

Based on USNRC regulatory guidance for DBA radiological consequence analyses, the U.S. EPR analyzes two iodine spiking cases for DBAs that do not result in fuel cladding damage. No other increase to activity is modelled, and for all cases the RCS noble gas activity concentration is assumed to be at the Technical Specification limit for DE Xe-133 of 210 µCi/gm.
Co-incident iodine spiking –
This case models the increase in RCS iodine activity concentration that would be expected to occur as a result of the primary system transient associated with the DBA. For the steam generator tube rupture (SGTR) accident, the RCS iodine activity concentration is assumed to be at the Technical Specification limit of 1.0 µCi/gm DE I-131 at the initiation of the accident, with addition of iodine for 8 hours at a rate of 335 times the equilibrium appearance rate (the rate at which iodine is released from the fuel to compensate for radioactive decay, leakage and clean-up to achieve the TS equilibrium concentration). For the main steam line break (MSLB) and small line break outside containment accidents, the RCS iodine activity concentration is assumed to be at the Technical Specification limit of 1.0 µCi/gm DE I-131 at the initiation of the accident, with addition of iodine at a rate of 550 times the equilibrium appearance rate for 8 hours.
The iodine appearance rates as listed in the U.S. EPR Design Control Document, Revision 5, are given in the table below.

Radionuclide	Equilibrium Appearance Rate (Ci/hr)
I-131	40.9
I-132	53.0
I-133	80.4
I-134	68.8
I-135	67.4

Pre-incident iodine spiking –
For the SGTR, MSLB and small line break outside containment, the RCS iodine activity concentration is assumed to be 60 µCi/gm DE I-131 at the initiation of the accident. This assumption models a case where a reactor transient has occurred prior to the DBA and has raised the RCS coolant iodine concentration above the technical specification limit.

8. REGULATORS' SOURCE TERMS ASSESSMENT CONCLUSIONS

Conclusions of RCS source terms assessment performed by the different regulators are summarized in this chapter. Some of them are preliminary, technical exchanges being pursued in some countries.

Finland

Main conclusions are as follows:

- The third source term "TechSpec" concentrations of the primary coolant was required,
- Attention was requested to if any further decrease in cobalt content of materials in the primary circuit is possible.

U.K.

There is no single report which defines the assessment of the source term for UK EPR; instead assessment was performed in the context of a number of different assessment topics considered as part of the GDA (Generic Design Assessment) process. Full details are available in the UK EPR GDA Step 4 Assessment reports, particularly;

- Radiation Protection, Section 4.1. www.hse.gov.uk/newreactors/reports/step-four/technical-assessment/ukepr-rc-onr-gda-ar-11-025-r-rev-0.pdf
- Reactor Chemistry, Section 4.2.3. www.hse.gov.uk/newreactors/reports/step-four/technical-assessment/ukepr-rc-onr-gda-ar-11-024-r-rev-0.pdf
- Design Basis Faults, Section 4.3. www.hse.gov.uk/newreactors/reports/step-four/technical-assessment/ukepr-fsdbf-onr-gda-ar-11-020a-r-rev-0.pdf

A summary of the main conclusions from these assessments regarding the source term can be summarised as below.

Radiation Protection

- ONR's assessment considered radiation sources from two perspectives. The first was to assess the management of the source term information, and the second was to assess reductions in the source term through selection of materials associated with the primary coolant.

- The definition and appropriate use of the source term is an important stage in understanding and deriving the safety requirements of any nuclear activity. ONR and the Environment Agency recognised that there was some consistency between the source terms used in different assessment areas, but there were also some apparent inconsistencies, and it was not always obvious how consistency was intended to be maintained. ONR and the Environment Agency requested EDF and AREVA to provide information on the following points:

    - How the radioactive source term had been derived.
    - Justification for the overall suitability of the source term.
    - Details of assumptions that could significantly affect the source term.
    - Identification of assessments where the source term was used and how it was used.
    - How the source term had been used to ensure consistently across the assessment areas.
    - How the source term had been manipulated for use in each specific assessment area along with assumptions used.

- EDF and AREVA provided a response which provided a description of source term management and organisation in EDF and AREVA, explained the primary source term definition, and provided references on the primary source term in the UK EPR, the primary nuclide source term derivation within systems, specific activity concentration of nuclides in reactor building systems, and activity concentrations in a range of systems. The response also provided information on the use of the source term in different assessment areas. ONR and the Environment Agency considered the information supplied in the response and agreed that the evidence presented satisfied the regulatory expectations regarding derivation of the source term, identification of assessments where the source term was used, use of the source term consistently across assessment areas, and use of the source term in specific assessment areas.
- During maintenance and repair work activities, worker exposure to radiation is mainly due to activated corrosion product deposits within the primary circuit of the pressurised water reactor which make a major contribution to dose rates in the vicinity of systems and components. The reduction of contamination is therefore of prime importance. The selection of materials which results in lower levels of corrosion products capable of activation in the primary circuit, therefore, helps to reduce dose rates in the vicinity of systems and components and thereby reduce worker radiation exposure. Assessment of reductions in source terms and radiation doses arising from reductions in the use of cobalt,

silver and antimony were carried out. Evidence to demonstrate source term reduction through selection of materials associated with the primary coolant was considered and ONR was supported in its assessment on radiation sources by technical support contractors. A literature review of radiological protection and radioactive waste and decommissioning practices during the last 10 years of normal operation of pressurized water reactors provided useful benchmark information.

- ONR considered the information regarding reductions in the levels of cobalt, silver and antimony from the source term in the UK EPR, and concluded that the reductions incorporated in the design compared with previous plants appeared ALARP and therefore further reductions of these elements from materials associated with the primary coolant were not necessary. Nevertheless, the restriction of exposure through material selection is dependent on procurement procedures. In addition, new materials may be developed before a UK EPR is constructed, in which case it would be appropriate for a further review of materials to be undertaken before future procurement. ONR captured this requirement in an Assessment Finding (AF-UKEPR-RP-01).
- In ONR's opinion, the evidence to substantiate the arguments relating to radiation sources regarding information on the source term, and reductions in the source term through selection of materials associated with the primary circuit, was suitable and sufficient.

Reactor Chemistry

- The reactor chemistry assessment considered several aspects related to the source term, including the influence of primary circuit chemistry and materials on radioactivity and the suitability of the source terms used in accidents.
- EDF and AREVA have provided detailed information on how the material and chemistry choices in UK EPR are predicted to influence the plant radiation fields, however most of this information is based on Operating experience feedback (OEF) from other reactors and not on analysis specific to UK EPR. Specifically the data for the radiochemical performance of UK EPR was not distinguishable from any other PWR made of similar materials. Since the radiochemistry of a reactor also depends on its size, power and other factors, a comparative review as theoretical calculations from internationally recognised experts in these fields have been commissioned. EDF and AREVA have provided design data to support these calculations and recently completed their own analysis for UK EPR. The calculations suggest that surface activities may be similar to existing plants if managed correctly. Total activity may be slightly higher than existing plants but this should be proportional to the power of the reactor. The significance of zinc, StelliteTM, boiling and surface areas have been identified. Parameters will be identifiable for subsequent stages of the project. These key parameters include: the release rate from steel surfaces in the reactor and the effect of zinc, and the level of dissolved (and particulate) corrosion product at the start of cycle.

- Overall, UK EPR follows the well-established and developed approach of restricting the material in contact with the primary coolant to mainly austenitic stainless steels (or cladding) or Ni-Cr-Fe alloys. EDF and AREVA have specified restrictive levels for impurities in these alloys and have described how the important factors such as conditioning and surface treatments will be specified to ensure releases are effectively controlled. ONR was satisfied with the material choices for UK EPR and content that EDF and AREVA have made an adequate ALARP argument for UK EPR.
- ONR noted that EDF and AREVA have taken great strides in removing cobalt alloys from wetted CVCS components which is a positive benefit. EDF and AREVA note the care needed in removing proven materials and state extensive test work to support this approach. They also intend to eliminate the antimony and silver that was used in ancillaries of the 1450 MWe French reactor.
- Overall, EDF and AREVA have presented reasonable arguments that cobalt radioactivity in UK EPR will be significantly lower than in earlier generations of PWR made with different materials.
- A number of contaminants (such as silver (by activation) or chloride (by corrosion)) can increase the radioactivity produced by any reactor and strict controls should be developed to prevent their ingress at all stages from fabrication, through commissioning to operations.

Design Basis Faults

- Through the assessment of Chapter 14.6 of the PCSR, ONR considered that it should be possible for future site specific analysis of design basis faults to show compliance with Target 4 of the SAPs (UK Safety Assessment Principles).
- While new site specific calculations have always been envisaged, ONR has still raised an Assessment Finding for site specific design basis radiological consequences analysis to be performed, taking due cognisance of usual UK methodology assumptions and explicitly comparing the results against Target 4 (***AF-UKEPR-FS-28***). It is acceptable to continue to use assumptions derived from either the “German” or “French” methodologies but these have to be justified on a case-by-case basis as being appropriate for the UK.

These assessments resulted in the following Assessment Findings related to the source terms which need to be addressed, as normal regulatory business, by the Licensee, during design, procurement, construction or commissioning phase of the new build project:

> ***AF-UKEPR-RP-01*** *- The licensee shall provide procurement procedures that require a review of materials associated with the primary coolant before purchase of those materials from their supplier in order to identify if there are any improvements in reductions in levels of cobalt or any other elements in materials which might lead to further reductions in radiation exposure of workers, and which would not compromise the functionality of*

those materials. This shall be complete before mechanical, electrical and control and instrumentation systems are delivered to site.

AF-UKEPR-RC-07 *– The Licensee shall ensure that a complete and unambiguous specification exists for all the materials to be used in UK EPR that could contact primary coolant. This should include trace elements prone to activation and be sufficiently detailed to allow sound procurement specifications to be produced. This Assessment Finding should be completed before such materials are delivered to site, but certain aspects may need earlier consideration, for example, to ensurerigorous control during procurement activities. Target milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components – delivery to Site.*

AF-UKEPR-RC-08 *– The Licensee shall ensure there is sufficient control over fabricators and operators that install, commission and maintain any hard-facing materials, including lapping, that may give rise to 60Co dose. This Assessment Finding should be completed before operations creating loose cobalt may take place on site, but certain parts may be necessary earlier, for example during component manufacture. Target milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components – delivery to Site.*

AF-UKEPR-RC-09 *– The Licensee shall review and consider alternative materials to Stellite™ for applications within UK EPR, and ensure that the final selection of materials is ALARP in this respect. This Assessment Finding should be completed before such materials are delivered to site for installation. Target milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components – delivery to Site.*

AF-UKEPR-RC-10 *– The Licensee shall keep the specification of secondary neutron sources under review and consider suitable alternatives. This Assessment Finding should be completed before reactor operations. Target Milestone – Initial criticality.*

AF-UKEPR-RC-11 *– The licensee shall define a surveillance programme for control rods and secondary neutron sources. The programme shall prevent the release of materials such as tritium and antimony before there is significant contamination of vessels or pipework. This Assessment Finding should be completed before nuclear operations. Target Milestone – Initial criticality.*

AF-UKEPR-RC-12 **–** *The Licensee shall generate evidence to support the lifetime behaviour of the nickel plating to be adopted for the pressuriser heaters in UK EPR. This should include consideration of material losses from the plating on radioactivity. This Assessment Finding should be completed before installation of the pressuriser is complete. Target milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - delivery to Site.*

AF-UKEPR-RC-13 *- The Licensee shall conduct sensitivity analysis for fuel crud formation in UK EPR. This should be used to demonstrate that levels of crud can be controlled and reduced So Far As Is Reasonably Practicable (SFAIRP) in UK EPR and should be based upon the detailed operating chemistry and core design for the UK EPR reactor. These calculations should provide balanced predictions of activity levels that allow the assessment of control measures including boiling patterns and StelliteTM replacements, as well as the management of significant chemicals and radionuclides. The Licensee shall conduct analyses of sensitivity to factors such as pH, zinc, boiling and dissolved corrosion* *products on crud build-up. The analysis should be used to justify related limits, conditions and criteria. This Assessment Finding should be completed before nuclear operations, as this is when fuel crud is formed. Target milestone – Initial criticality.*

AF-UKEPR-RC-25 *– The Licensee shall specify the acceptable level for tritium in the Spent Fuel Pool and connected systems, including the IRWST. This should include evidence that operator radiation exposure and discharges have been considered. This Assessment Finding should be completed before nuclear operations, as this is when tritium will first be generated. Target Milestone – Initial criticality.*

AF-UKEPR-FS-28 *- The future licensee shall provide site specific radiological consequences analysis for design basis events (including hazards), taking due cognisance of usual UK methodology assumptions and explicitly comparing the results against Target 4. Single failure assumptions and sensitivity cases should be reviewed and addressed on their merits for the UK.*

In addition the assessment identified a related Reactor Chemistry GDA issue which requires resolution:

GI-UKEPR-RC-02 – Control and Minimisation of Ex-core Radiation *– EDF and AREVA to demonstrate that ex-core radiation levels in UK EPR are minimised so far as is reasonably practicable and can be controlled.*

France

Main conclusions are as follows:

- Reactor coolant activities defined by EDF may be used as inputs for the different chapters of the SAR related to plant normal release, radiation protection, and description of realistic dose rates, inside and outside the containment…
- It was recommended that the third source term be at least envelope of activities obtained in all French NPP. EDF answered that this source term encompasses the operating feedback of 1300 and N4 NPP between 1990 and 2004 (time period considered for the assessment) and of all French NPP between 1995 and 2004. IRSN has to analyze this answer.

Regarding activated corrosion products:

- Main isotopes to be considered in corrosion products have been identified by EDF but the list provided by EDF is not fully exhaustive. In particular, Niobium is not considered.
- Source term in Co-60 should be significantly reduced regarding plants in operation with the reduction of the use of stellites but it has not been completely eliminated (main coolant pumps, control rods mechanisms…). Therefore, EDF has to justify that it is not possible (or not in accordance with ALARA/ALARP considerations) to eliminate stellites completely.
- The improvement of steam generator tubes processes plays an important role in the decrease of Co-58. But recent feedback on French NPPs after steam generator

replacement shown relative high level of Co-58 during first cycles. EDF should consider margins to cover the lack of clear understanding of some physical and chemical phenomenon.

- Ag-110m (silver): EDF should define a plan for checking and maintain control rods in order to avoid a loss of leaktighness. EDF should also justify the use of Helicoflex joints with Ag.
- Activity peak during oxygenation: EPR values are not envelope of activities observed on some 900 MWe plants. EDF is asked to take into account the feedback on these plants and takes margins to cover the lack of understanding of phenomenon taking place.
- Ni-63: it is 15 to 20% of the activity released in the environment. It cannot be measured by routine sensors. Only two campaigns of measurements have been performed on site to evaluate its behaviour in the installation. EDF should perform new campaigns to have a better understanding of its behaviour.
- Concerning activity deposit inside pipes, EDF is asked to consider more radionuclides (as a minimum, Cr-51, Zr-95 and Nb-95) and to check that the main coolant pipes are the most radioactive parts.

Regarding coolant activation products:

- Ar-41: values for EPR are not envelope of values measured in existing plants. Production of Ar-41 depends on operating modes. EDF should review the activity in Ar-41 taking into account the feedback of operating plants (and not only N4) and EPR operating modes. EDF considers that some values are false. EDF should justify this point.
- C-14: several campaigns have been performed in existing plants but the reliability of measurements has been questioned. EDF is encouraged to pursue its study on the behaviour of C-14 in the main coolant before defining the C-14 source term.
- Tritium: a value of 37 GBq/t has been proposed by EDF for all source terms. It covers feedback on 1300 MWe and N4 plants and it has been judged acceptable by IRSN. ASN asked EDF to check that the impact of cladding defects on H-3 production will not question the RCS H-3 volume activity taken into account for Flamanville 3 EPR.

Regarding fission products:

- The methodology for existing plants and EPR differs; EDF was asked to clarify its approach for fission products source term assessment.
- Several errors and inconsistencies have been detected – EDF has been asked to correct the values (inconsistencies in data used for statistical analysis of feedback).
- ASN has estimated that the values for EPR should be envelope of all existing plants in France.

As a conclusion, ASN requested EDF to assess the impact of its recommendations on the nuclear pressure equipment classification (as it depends on the activity of the fluid in the circuits).

U.S.

The U.S. EPR is currently under review. The USNRC's conclusions on the source terms for the U.S. EPR will be documented in the Final Safety Evaluation Report (FSER), to be published as a NUREG report upon completion of the staff's review. The FSER will be publically available on the USNRC website. Specific information on the USNRC's evaluation of the U.S. EPR source terms and related analyses will be found in chapters 11, 12 and section 15.0.3 of the FSER.

China

Under review

9. CONCLUSION

This report gives information on the way EPR RCS source term is elaborated and used in China, US, UK, Finland and France at the design stage. It identifies main discrepancies and their origin. Discrepancies are not really linked to the EPR design but on historical practices, feedback available and different used methods.

It can be underlined that EDF has taken into account the conclusions of the first RCS source terms assessment performed in France for the Flamanville 3 EPR in the files submitted in UK and China, which is satisfactory.

If discussions are still on-going in some countries concerning the extent of the operating experience feedback to be considered to determine EPR RCS source term (types of plant, percentage of coverage, case-by-case justifications…), several regulators have recognized efforts made by the designer to remove cobalt alloys from RCS equipment.

To limit the source term, operators should then take care of operating procedures.

The behavior of some products is still uncertain and operators are encouraged to pay a particular attention to RCS source terms in the first years of operation to evaluate the effective reduction of RCS source terms regarding existing plants, especially French and German latest series. The operating feedback could be then taken into account for future projects.

As a conclusion, although it seems difficult to share common position within the MDEP EPRWG on this issue, recommendations may then be provided on good practices, for instance SG electro-polishment, RCS surface passivation, zinc injection… However, some of these practices may be unique to individual EPR plants and not EPR design-related.

10. REFERENCES

[1] 6th EPR Working Group meeting – May 2010 - minutes

[2] EPRWG radiation protection experts meeting – September 2010 - minutes

[3] ESPN order published in December 2005 related to Nuclear Pressure Equipment - France

[4] "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors", NUREG-0017, Rev 1, US Nuclear Regulatory Commission, Washington, D.C. 1985)

[5] NRC Website for New reactors - http://www.nrc.gov/reactors/new-reactors.html

[6] NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (NUREG-0800, Formerly issued as NUREG-75/087) - http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/

[7] NRC Regulatory Guide 8.8: "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable"

[8] ANSI/ANS-18.1-1999: Radioactive Source Term for Normal Operation of Light Water Reactors - http://www.ans.org/store/i_240238

[9] 10 CFR Part 20, appendix B - Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage - http://www.nrc.gov/reading-rm/doc-collections/cfr/part020/part020-appb.html