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REVISION HISTORY

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01	Inclusion of BAT forms for iodines and noble gases (sections 3.5 and 3.6) Minor changes in section 2 (table 2) and in section 3.2	30-10-2009
02	Minor editorial changes	09-11-2009
03	Relevant references marked as CCI Minor editorial changes Complementary information included in sections - 3.2.5, - 3.3.3.1, 3.3.5.2, - 3.4.3.1.1, 3.4.3.1.3, 3.4.4.1, - 3.5.3, 3.5.4.1, - 3.6.4.2, 3.6.5.1.	28-03-2010
04	PCER March 2011 update - Minor editorial corrections - Information added regarding choice of materials and primary coolant chemistry. - Information added regarding assessment for the filtration and demineralisation systems of the CSTS, LWPS and SGBS systems.	30-03-2011
	Protect Commercial marking removed	
05	Reformatted PCER and Supporting Document Version - Minor formatting changes and editorial corrections	31-05-2012

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REVISION HISTORY (Cont'd)

Issue	Description	Date
06	Consolidated PCER and Supporting Document update - Glossary updated - Alignment with PCER Chapters 5.5, 6.3, 8.2 and 11 and cross-references added - Reference updated - Clarification in Table 2, sections 3.1, 3.2, 3.3, 3.4, 3.5, 3.6 and 5 - Minor editorial changes in sections 3.2, 3.3, 3.5 and 3.6 - BAT demonstration of zinc injection included	21-08-12

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SUMMARY

This report has been prepared to provide a key reference in the GDA UK EPR Integrated Waste Strategy document [UKEPR-0010-001].

The objective of the report is to provide documentary evidence that the Best Available Techniques (BAT) have been used:

- to prevent, and where that is not practicable, to minimise the production of radioactive waste at source; and
- to ensure that the overall environmental risk and impact are minimised as a whole, in the management of radioactive waste.

The information presented is intended to provide assurance that the EPR design meets the requirement for application of BAT. Much of the information presented has been drawn from the UK EPR Pre-Construction Environmental Report (PCER) but other information has also been used, including international literature relating to waste management and detailed technical reports.

The evidence provided in this report includes the following information:

- the identification of radionuclides that can be produced in the facility in significant quantities and that will be present in radioactive wastes (solid, liquid and gaseous) for disposal;
- the mechanisms for production of the identified radionuclides and the source of arisings in terms of process streams and equipment;
- the quantification of the annual production at source of each identified radionuclide in terms of activity;
- the techniques used to prevent or minimise the amount of activity produced at source and for abatement; and
- the assessment of the significance of each radionuclide with respect to either the activity generated, the total activity disposed of to the environment or the potential impact in terms of radiation dose to humans.

Information which addresses these aspects is presented in tabular format. A small number of radionuclides have been assessed as significant, namely carbon-14, tritium, cobalt-58 and cobalt-60, noble gases and iodine isotopes (noting that isotopes of cobalt can be considered to be broadly representative of other radionuclides arising as a result of corrosion processes that are relevant to radioactive waste discharge and disposals). For the specific radionuclides identified more detailed information has been provided in "BAT forms". These contain information on:

- the mechanisms of production of the radionuclide and the process by which it is produced;
- the source term including, where possible, information on the activity before and after the application of techniques to prevent or minimise at source;

- the demonstration that the techniques used to minimise at source are BAT by reference to all other options considered or rejected or by benchmarking against best world practice;
- the demonstration that the techniques used for treating each process stream minimise the overall impact of the disposal and discharge of radioactive waste (including evidence of the consideration of disposal via different pathways);
- the demonstration that the techniques used to minimise impacts are BAT by reference to all other options considered and rejected or by benchmarking against best world practice;
- the demonstration that the techniques used conform to the Integrated Waste Strategy (IWS) (which is based on the Flamanville 3 Reference Case of the UK EPR GDA design submission) by cross-referencing to the relevant IWS sections; and
- the identification of disposals resulting from the use of techniques to minimise overall impacts and quantification (where possible) of annual volumes and activity of nuclides.

This is considered appropriate as routine discharges of these radionuclides are likely to be limited by the Environment Agency and their magnitude is indicative of effective reactor and abatement plant operation.

The information presented demonstrates that the Best Available Techniques are being applied in the design of the EPR to minimise radioactive wastes at source and to minimise the impacts of the disposal of wastes into the environment.

The report covers the radioactive wastes arising from reactor operations, noting that the optimised EPR design will influence decommissioning waste management requirements. Although the primary focus of demonstration of BAT is on wastes arising from reactor operations, the information presented is also of relevance to future arisings of decommissioning wastes.

GLOSSARY

AECL	Atomic Energy of Canada Limited
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
APG [SGBS]	Steam Generator Blowdown System
BAT	Best Available Technique
BPEO	Best Practicable Environmental Option
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium
CfA	Conditions for Acceptance
DWN [NABVS]	Nuclear Auxiliary Building Ventilation System
CVI	Condenser Vacuum
EBA [CSVS]	Containment Sweep Ventilation System
EDF	Electricité De France
EFPD	Effective Full Power Days
EPR	European Pressurised Water Reactor
EPRI	Electric Power Research Institute
ETB	Effluent Treatment Building
EVF	RB Internal Filtration System
EVR [CCVS]	Containment Cooling and Ventilation System
IER	Ion Exchange Resin
FB	Fuel Building
GDA	Generic Design Assessment
HEPA	High Efficiency Particulate Air
HSE	Health & Safety Executive
HVAC	Heating Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
IER	Ion Exchange Resin
ILW	Intermediate Level Waste
INPO	Institute of Nuclear Power Operations
IRWST	In-containment Refuelling Water Storage Tank
IWS	Integrated Waste Strategy
0KER [LRMDS]	Liquid Radwaste Monitoring and Discharge System
KRT [RPMS]	Radiological Protection Measurement System
LCO	Limit Condition of Operation
LLW	Low Level Waste
LLWR	Low Level Waste Repository
LWR	Light Water Reactor
NAB	Nuclear Auxiliary Building
NEI	Nuclear Energy Institute
OECD	Organisation for Economic Cooperation and Development
OEF	Operating Experience Feedback

OPEX	Operational Experience
PCER	Pre-Construction Environmental Report
PCSR	Pre-Construction Safety Report
PTR [FPC(P)S]	Fuel Pool Cooling (and Purification) System
PWR	Pressurised Water Reactor
RB	Reactor Building
REA [RBWMS]	Reactor Boron and Water Make-up System
RCP [RCS]	Reactor Coolant System
RCV [CVCS]	Chemical and Volume Control System
RSA'93	Radioactive Substances Act 1993
RPE [NVDS]	Nuclear Vent and Drain System
RWMC	Radioactive Waste Management Case
SB	Safeguard Building
SEK [CILWDS]	Conventional Island Liquid Waste Discharge System
TEG [GWPS]	Gaseous Waste Processing System
TEP [CSTS]	Coolant Storage and Treatment System
TES [SWTS]	Solid Waste Treatment System
TEU [LWPS]	Liquid Waste Processing System
TSS	Totally Suspended Solid
VCT	Volume Control Tank
VLLW	Very Low Level Waste

1. INTRODUCTION

This report has been prepared to provide a key reference in the GDA UK EPR Integrated Waste Strategy document [UKEPR-0010-001].

The objective of the report is to provide documentary evidence that the Best Available Techniques (BAT) have been used:

- to prevent, and where that is not practicable, to minimise the production of radioactive waste at source; and
- to ensure that the overall environmental risk and impact are minimised as a whole, in the management of radioactive waste.

The information presented is intended to provide assurance that the EPR design meets the requirement for application of BAT. Much of the information presented has been drawn from the UK EPR Pre-Construction Environmental Report (PCER) but other information has also been used, including international literature relating to waste management and detailed technical reports.

The evidence provided in this report includes the following information:

- the identification of radionuclides that can be produced in the facility in significant quantities and that will be present in radioactive wastes (solid, liquid and gaseous) for disposal and environmental discharges;
- the mechanisms for production of the identified radionuclides and the source of arisings in terms of process streams and equipment;
- the quantification of the annual production at source of each identified radionuclide in terms of activity;
- the techniques used to prevent or minimise the amount of activity produced at source and for abatement; and
- the assessment of the significance of each radionuclide with respect to either the activity generated, the total activity disposed of to the environment or the potential impact in terms of radiation dose to humans.

Information which addresses these aspects is presented in tabular format. A small number of radionuclides have been assessed as significant on, namely carbon-14, tritium, cobalt-58 and cobalt-60, noble gases and iodine isotopes (noting that isotopes of cobalt can be considered to be broadly representative of other radionuclides arising as a result of corrosion processes). For these specific radionuclides more detailed information has been provided in “BAT forms”, which are provided in section 3.

Section 4 provides information on the radionuclides that are proposed to be individually limited and those which are proposed to be grouped for the purpose of setting disposal limits.

BAT forms are included for iodine isotopes and noble gases, which were identified to be of medium impact in the assessment (section 2). This is considered appropriate as routine discharges of these radionuclides are likely to be subject to limits and their magnitude is indicative of effective reactor and abatement plant operation.

The information in the BAT forms is largely focused on minimisation at source and minimisation of the impacts of disposal to the environment associated with gaseous and liquid discharges. Operational arisings of solid wastes from reactor operations are primarily associated with the application of techniques to minimise the impacts of disposal of gaseous and liquid discharges, for example, filtration to remove particulate material from liquid and gaseous discharges and ion exchange processes which remove soluble material from liquid discharges. Further information on solid waste arisings is presented in the IWS document and in more detail in Sub-chapter 6.3 of the PCER and other supporting references.

The information presented in the BAT forms is based largely on the radioactive wastes arising from reactor operations, noting that the EPR design will be a key factor in determining decommissioning waste management requirements. Although the primary focus of demonstration of BAT is on wastes arising from reactor operations, the information presented is of relevance to future arisings of decommissioning wastes. Minimisation of wastes at source and minimisation of the impacts of disposal by means of effective abatement techniques are key factors which will minimise future decommissioning wastes. The principles underlying the future decommissioning of the EPR are discussed in Chapter 5 of the PCER and in the IWS document.

1.1 MAPPING OF INFORMATION IN THE REPORT

Table 1 provides information to indicate where evidence can be found in this report.

Table 1: Mapping of Information in the Report

Evidence required	Location of information in document and comments
Application of Best Available Techniques (minimisation at source)	
Identify radionuclides that can be produced in the facility in significant quantities (in terms of radioactivity) and that will be present in radioactive wastes (solid, liquid and gaseous) for disposal.	Table 2, section 2.1. Information on the source terms is presented for all nuclides in Table 2 while additional information on the source term for significant radionuclides is presented in Part 2 of the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6.
Describe the mechanisms for production of each identified radionuclide and relate it to the process stream, equipment, item or facility area where produced.	Table 2, section 2.1 for all radionuclides and Part 1 of the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6 for significant radionuclides.
For each identified radionuclide describe the techniques used to prevent or minimise the amount of activity produced at source.	Summary information presented in Table 2, section 2.1 and Part 3 in sections 3.2, 3.3, 3.4, 3.5 and 3.6 for significant radionuclides.
For each identified radionuclide quantify the typical reduction in activity that will be achieved by each technique used.	No specific information is presented for all identified radionuclides. Where information is available for significant radionuclides it is presented in Part 2 and other parts of the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6. In practice it is very difficult to quantify the reduction in activity achieved as a result of minimisation at source, because of difficulties in quantifying the primary source terms where mechanisms of formation are complex.
Demonstration that the techniques used to minimise activity at source are BAT by reference to all other options considered or rejected or by benchmarking against best world practice.	Information is presented for significant radionuclides in Part 3 of the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6.
Quantify the annual production at source of each identified radionuclide in terms of activity.	Table 2, section 2.1 and in Part 2 of the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6.

Table 1: Mapping of Information in the Report (continued)

Evidence required	Location of information in document and comments
Assess the significance of each radionuclide with regard to either the activity generated, or the total activity disposed of to the environment, or the potential impact in terms of dose to humans (i.e. the critical group). Radionuclides that have an insignificant impact should be documented but need not be considered in the management and abatement assessments. However for significant radionuclides you should: • Identify radionuclides that you propose should be individually limited. • Identify radionuclides that you propose should be grouped together for setting disposal limits.	Table 2, section 2.1 for assessment of significance. Section 4 for proposals with respect to individual limits and groups of radionuclides for the purpose of setting disposal limits.
List and justify any assumptions underpinning the BAT assessment.	No specific list prepared but assumptions and any associated justifications are presented in the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6. References are provided to support statements made in the BAT forms wherever possible, primarily to the Pre-Construction Environmental Report but also to other information sources.
Application of Best Available Techniques (management and abatement)	
Demonstration that the techniques used for treating each process stream minimise the overall impact of the disposal of radioactive waste (including evidence of the consideration of disposal via different pathways).	Part 4 of the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6.
Demonstration that the techniques used conforms to the Integrated Waste Strategy (IWS) by cross-referencing to the relevant IWS sections.	Part 6 of the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6.

Table 1: Mapping of Information in the Report (continued)

Evidence required	Location of information in document and comments
Identification of disposals resulting from the use of techniques to minimise overall impacts and quantification (where possible) of annual volumes and activity of nuclides.	Part 7 of the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6.
Demonstration that the techniques used to minimise impacts are BAT by reference to all other options considered and rejected or by benchmarking against best world practice.	Part 5 of the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6.
List and justify any assumptions underpinning the BAT assessment.	No specific list prepared but assumptions and any associated justifications are presented in the BAT forms in sections 3.2, 3.3, 3.4, 3.5 and 3.6. References are provided to support statements made in the BAT forms wherever possible, primarily to the Pre-Construction Environmental Report but also to other information sources.
Radioactive waste discharges – Carbon-14	
The provision of a BAT assessment for the prevention or minimisation of discharges of carbon-14 in liquid and gaseous radioactive waste. This should as a minimum include: • options in handling process streams to optimise discharges between gaseous and liquid waste, e.g. use of evaporators; • options to abate gaseous waste by use of conversion to carbon dioxide and subsequent scrubbing.	Section 3.2 of this report (the BAT form for carbon-14), most in particular in Parts 4 and 5.
A review which justifies the proposed emission limit values for carbon-14 in liquid and gaseous waste discharges in the light of the outcomes of the BAT assessment.	Partially addressed in section 3.2 of this report (the BAT form for carbon-14), most in particular in sections 2, 5 and 7. Further information is presented in Sub-chapter 6.3 of the PCER.

2. IDENTIFICATION OF RADIONUCLIDES, SOURCE TERMS AND ASSESSMENT OF SIGNIFICANCE

Concise information on the radionuclides that will arise as a result of operation of the EPR has been compiled in Table 2. For each radionuclide [ENTERP090062 A EPR Primary Source Term of the EPR reactor, EDF, March 2009] information is provided on:

- half-life;
- source (radionuclide) and source term (in MBq t^{-1} and total source term based on number of tonnes of fluid in the primary circuit where applicable);
- mechanism and source of production of radionuclide;
- whether the radionuclide is an indicator of plant performance;
- techniques used to minimise activity at source;
- techniques used to minimise the impacts of disposals (abatement, etc.).

The information in Table 2 was used to assess the significance of radionuclides. The following four factors were taken into account in assessing significance with respect to preparation of nuclide specific BAT forms, and which are considered to be based on a pragmatic balanced view of relevant factors for which no statutory guidance exists:

1. half-life (there is no specific threshold value for significance of half-life);
2. magnitude of source term (where significant is defined as a large source term, i.e. number of becquerels);
3. contribution to radiation doses (where significant is defined as > 5% of total dose for either liquid or gaseous discharges, for individual radionuclides, for the maximally exposed group, based on Sub-chapter 11.1 of the PCER);
4. whether the radionuclide is an indicator of plant performance.

Table 2 contains information on:

- significance;
- reasoning for assessment of significance;
- whether a BAT form is required.

The nuclides for which BAT forms have been prepared are highlighted in yellow in Table 2 where the significance is assessed as low, medium or high. This assessment has been made by taking into consideration the four factors mentioned above.

BAT forms have been prepared for all radionuclides identified as having a high significance (carbon-14, tritium (H-3) and isotopes Co-58 and Co-60), all of which are significant in terms of radiation dose (noting that Co-58 is grouped with Co-60 because of its chemical similarity not because of its significance alone). Carbon-14 and tritium are also significant because of the magnitude of discharges in numerical terms. The BAT forms are presented in section 3.

Several other nuclides were identified as being of less significance (Iodine-131, Caesium-134 and Caesium-137 and Xenon-133) because they have significant source terms or are indicators of plant performance but they do not give rise to significant radiation doses. As noted in section 1, BAT forms have been developed for iodine isotopes and noble gases to cover the issues arising from their consideration.

Table 2: Table of radionuclides arising from operation of the EPR

The nuclides for which BAT forms have been prepared are highlighted in yellow, where the significance is assessed as low, medium or high by taking into consideration the four factors mentioned above.

Radio-nuclide	Half-life	Source	Source term in MBq t^{-1}	Source Term MBq (unless stated otherwise)	Mechanism and source of production of radionuclide (fission, activation or corrosion product)	Reasoning for assessment of significance (based on half-life, magnitude of source term or discharges, dose impact or as indicator of plant performance)	Significance	BAT Form Required?	Techniques used to minimise activity produced at source	Techniques used to minimise impacts of disposal
Ag-110m	249.9 d	Ag-109	3.2	945.92	Corrosion product. Ag-109 is present in silver-containing seals and is activated to form Ag-110m. Corrosion results in its presence in primary coolant [Reference: Radiochemistry of Nuclear Power Plants with Light Water Reactors, K-H. Neeb, de Gruyter, 1997]. It is also present in the control rods (with indium and cadmium, see PCER Sub-chapter 6.1, section 2.6).	Relatively short half-life (< 1year), low magnitude in source term and is not a significant contributor to radiation doses.	Low	No	Minimisation of sources that could give rise to Ag-110m, e.g. reduction in the use of helicoflex seals in favour of graphite seals. Reinforced leak tightness requirements for active parts, pH control to minimise corrosion (see section 3.3.1 of Sub-chapter 8.2 of the PCER). It is not possible to abate the Ag-110m arising from the control rods.	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER).
Ar-41	1.8 h	Ar-40	300	88680	Activation. Argon-41 (another radioactive noble gas) is formed during normal operation by activation of the natural content of Argon-40 in the air around the reactor pressure vessel by the neutron radiation, in the Reactor Building. The discharges of Ar-41 are associated with the start-up of the EBA [CSVS] prior to unit shutdown, and if access is required during operational phases. The discharge then occurs from the Reactor Building rather than from the TEG [GWPS] (section 7.4.2 of Sub-chapter 6.3 of the PCER).	Short half-life (< 1d) and is not a significant contributor to radiation doses.	Low	No	No practicable means of minimisation at source as it is formed as a result of the presence of Ar-40 in air.	Its half-life is under two hours and it therefore appears only transiently and in circumstances of Reactor Building venting. Ar-41 is collected by the ventilation of the Reactor Building (EBA [CSVS]), and discharged to atmosphere when the ventilation is operational (section 7.4.2 of Sub-chapter 6.3 of the PCER). Ventilation system design ensures that the impact of disposals are minimised.
C-14	5730 y	O-17, N-14 and C-13	6	518 GBq y^{-1}	Activation. See BAT form for detailed information on mechanism and source of formation.	Long half-life (>>> 1 year). Magnitude of source term is large and is the major contributor to radiation doses (collective dose, dose to maximally exposed individuals and non-human biota) from gaseous and liquid discharges.	High	Yes	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
Co-58	70.78 d	Ni-58	21	6207.6	Corrosion. See BAT form for detailed information on mechanism and source of formation.	Relatively short half-life (< 1 year) and is not a significant contributor to radiation doses.	Low	Yes (with Co-60)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
Co-60	5.27 y	Co-59	2.3	679.88	Corrosion. See BAT form for detailed information on mechanism and source of formation.	Relatively long half-life (> 1 year) is significant contributor to doses to maximally exposed individuals and non-human biota for liquid discharges and is an indicator of plant performance (corrosion).	High	Yes	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.

Table 2: Table of radionuclides arising from operation of the EPR (continued)

Radio-nuclide	Half-life	Source	Source term in MBq t^{-1}	Source Term MBq (unless stated otherwise)	Mechanism and source of production of radionuclide (fission, activation or corrosion product)	Reasoning for assessment of significance (based on half-life, magnitude of source term or discharges, dose impact or as indicator of plant performance)	Significance	BAT Form Required?	Techniques used to minimise activity produced at source	Techniques used to minimise impacts of disposal
Cr-51	27.7 d	Cr-50	28	8276.8	Corrosion. Formed as a result of neutron activation of Cr-50. Chromium is present in chromium steel used in structural components [Reference: Radiochemistry of Nuclear Power Plants with Light Water Reactors, K-H. Neeb, de Gruyter, 1997].	Relatively short half-life (<< I year), low magnitude in source term and no significant contribution to radiation doses.	Low	No	In order to limit the source term, particular care has been given to the specification of all materials in contact with the primary coolant in the primary circuit. Control of the source term is achieved through careful selection of materials in contact with the primary coolant which then leads to a reduction in the production of corrosion products [NEEM-F DC 140 Revision A Reduction in primary circuit radioactivity SFAIRP based upon the primary circuit materials]. In addition, hot functional tests are carried out, in which contact of the primary circuit with water at high temperature produces surface corrosion and an oxide layer on its surfaces [NEEM-F DC 143 Revision B, “Reduction in Primary Circuit Radioactivity SFAIRP Based Upon the Primary Circuit Chemistry]. The less general corrosion, the lower source term will be and the more protective the layers, the less corrosion particles leakage will occur during cool-down periods. Operating Experience Feedback from EDF-N4 units, Sizewell B and Tomari 3 suggests that a good passive film has been produced throughout the primary circuit, minimising corrosion product release. In addition careful control of the coolant chemistry minimises corrosion (see section 3.3.1 of Sub-chapter 8.2 of the PCER). Much of the information in the BAT form for the reduction of Co-58 and Co-60 is relevant to all corrosion products. In addition, the zinc injection has a supplementary effect on cobalt inhibition and replacement.	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER).

Table 2: Table of radionuclides arising from operation of the EPR (continued)

Radio-nuclide	Half-life	Source	Source term in MBq t^{-1}	Source Term MBq (unless stated otherwise)	Mechanism and source of production of radionuclide (fission, activation or corrosion product)	Reasoning for assessment of significance (based on half-life, magnitude of source term or discharges, dose impact or as indicator of plant performance)	Significance	BAT Form Required?	Techniques used to minimise activity produced at source	Techniques used to minimise impacts of disposal
Cs-134	2.06 y		40	11824	Fission of uranium. Cs-134 is not a direct fission product but is created in the fuel by the activation of Cs-133 (which is a direct fission product). Fission products are usually present in the reactor cooling water. Despite a high standard of cleanliness a trace of uranium always remains on fuel surfaces after the manufacturing process. Once the fuel is in the reactor, this “tramp” uranium will fission, producing fission products in the reactor cooling water. Another route for fission products to enter the reactor coolant system is as a result of fuel leaks (section 6.4.2 of Sub-chapter 6.3 of the PCER).	Relatively long half-life (>1 year) but source term is relatively low and is not a significant contributor to radiation doses. Isotope is an indicator of fuel performance issues.	Medium	No	Fission products are minimised as a result of a reduction in the amount of natural uranium relative to the amount of energy produced, in comparison with other reactor designs. Examples of measure include better yield from fuel as a result of improved design and performance. The design of the EPR incorporates measures for the optimisation of fuel use as a result of the "large core" and use of the steel reflector, which reduce neutron leakage and increased thermal efficiency (section 3.2 of Sub-chapter 8.2 of the PCER).	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER).
Cs-136	13.16 d		3.7	1093.72	Fission of uranium, see also Cs-134.	Relatively short half-life (<< 1 year). Low magnitude in source term and no significant contribution to radiation exposure.	Low	No	Fission products are minimised as a result of a reduction in the amount of natural uranium relative to the amount of energy produced, in comparison with other reactor designs. Examples of measure include better yield from fuel as a result of improved design and performance. The design of the EPR incorporates measures for the optimisation of fuel use as a result of the "large core" and use of the steel reflector, which reduce neutron leakage and increased thermal efficiency (section 3.2 of Sub-chapter 8.2 of the PCER).	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER).
Cs-137	30 y		40	11824	Fission of uranium, see also Cs-134.	Relatively long half-life (>> 1 year) but source term is relatively low and is not a significant contributor to radiation doses. Isotope is an indicator of fuel performance issues.	Medium	No	Fission products are minimised as a result of a reduction in the amount of natural uranium relative to the amount of energy produced, in comparison with other reactor designs. Examples of measure include better yield from fuel as a result of improved design and performance. The design of the EPR incorporates measures for the optimisation of fuel use as a result of the "large core" and use of the steel reflector, which reduce neutron leakage and increased thermal efficiency (section 3.2 of Sub-chapter 8.2 of the PCER)	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER)

Table 2: Table of radionuclides arising from operation of the EPR (continued)

Radio-nuclide	Half-life	Source	Source term in MBq t^{-1}	Source Term MBq (unless stated otherwise)	Mechanism and source of production of radionuclide (fission, activation or corrosion product)	Reasoning for assessment of significance (based on half-life, magnitude of source term or discharges, dose impact or as indicator of plant performance)	Significance	BAT Form Required?	Techniques used to minimise activity produced at source	Techniques used to minimise impacts of disposal
Cs-138	32.2 min		850	251260	Fission of uranium, see also Cs-134.	Short half-life (<1d) in comparison with Cs-137 and is not a significant contributor to radiation exposure.	Low	No	As for Cs-134.	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER).
Fe-59	45.1 d	Fe-58	1.3	384.28	Corrosion. Formed by neutron activation of Fe-58. Iron is present in alloys used in reactor components.	Relatively short half-life (<< 1 year), low magnitude in source term and no significant contribution to radiation exposure.	Low	No	As for Cr-51.	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER).
H-3	12.33 y	B-10 and Li-6	37000	10937200	Activation. See BAT form for detailed information on mechanism and source of formation.	Relatively long half-life (>> 1 year). Magnitude of source term is large and it is a significant contributor to radiation doses (collective and maximally exposed individuals) from gaseous and liquid discharges.	High	Yes	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
I-131	8.04 d		100	29560	Fission. Iodine isotopes are formed in the fuel by fission and can escape into the reactor coolant water via fuel defects. Also, like other fission products, small quantities are produced from uranium contamination on fuel surface (“tramp” uranium) within the reactor which can also be found in the primary coolant. See also Cs-134.	Relatively short half-life (<< 1 year), but it is not a significant contributor to radiation doses from gaseous discharges, use of charcoal filtration system to minimise discharges.	Medium	Yes	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
I-132	2.3 h		190	56164	As for I-131.	Short half-life (< 1d), use of charcoal filtration system to minimise discharges.	Low	Yes (with I-131)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
I-133	20.8 h		310	91636	As for I-131.	Short half-life (< 1d), use of charcoal filtration system to minimise discharges.	Low	Yes (with I-131)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
I-134	52.6 min		190	56164	As for I-131.	Short half-life (< 1d), use of charcoal filtration system to minimise discharges.	Low	Yes (with I-131)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
I-135	6.61h		200	59120	As for I-131.	Short half-life (< 1d), use of charcoal filtration system to minimise discharges.	Low	Yes (with I-131)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.

Table 2: Table of radionuclides arising from operation of the EPR (continued)

Radio-nuclide	Half-life	Source	Source term in MBq t^{-1}	Source Term MBq (unless stated otherwise)	Mechanism and source of production of radionuclide (fission, activation or corrosion product)	Reasoning for assessment of significance (based on half-life, magnitude of source term or discharges, dose impact or as indicator of plant performance)	Significance	BAT Form Required?	Techniques used to minimise activity produced at source	Techniques used to minimise impacts of disposal
Kr-85	10.72 y		38	11,232.8	Fission. Radioactive noble gases are formed by fission. They are usually confined in the fuel but, in the event of fuel leaks, they can pass into the primary coolant via defects in the fuel cladding. Their presence in the primary coolant is also due to the occurrence of traces of uranium ("tramp" uranium) that can never be completely removed on new fuel assemblies following the manufacturing process. See also Cs-134.	Despite relatively long half-life it is not a significant contributor to radiation doses from atmospheric discharges and the magnitude of the source term is relatively low.	Low	Yes (with Xe-133)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
Kr-85m	4.48 h		200	59,120	As for Kr-85	Short half-life (< 1d), use of carbon-bed delay systems to minimise discharges.	Low	Yes (with Xe-133)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
Kr-87	1.27 h		360	106,416	As for Kr-85.	Short half-life (< 1 d).	Low	Yes (with Xe-133)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
Kr-88	2.84 h		500	147,800	As for Kr-85.	Short half-life (< 1 d).	Low	Yes (with Xe-133)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
Mn-54	312.5 d	Fe-54	4.2	1241.52	Corrosion. Formed by neutron activation of Fe-54. Iron is present in alloys used in reactor components.	Relatively short half-life (< 1 year). Low magnitude in source term and is not a significant contributor to radiation doses.	Low	No	As for Cr-51.	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER).
N-16	7.3 s	O-16			Activation of O-16 present in air, coolant and other substances present in the reactor.	Short half-life (< 1 d).	Low	No	No practicable means of minimisation at source as it is formed as a result of the presence of O-16.	Minimisation of impacts not necessary as a result of its short half-life.
N-17	4.2 s	O-17			Activation of O-17 present in air, coolant and other substances present in the reactor.	Short half-life (< 1 d).	Low	No	No practicable means of minimisation at source as it is formed as a result of the presence of O-17.	Minimisation of impacts not necessary as a result of its short half-life.
Ni-63	100 y	Ni-62	15	4434	Corrosion. Formed by neutron activation of stable Ni-62 present in structural materials.	Low magnitude in source term and is not a significant contributor to radiation doses.	Low	No	As for Cr-51	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER).

Table 2: Table of radionuclides arising from operation of the EPR (continued)

Radio-nuclide	Half-life	Source	Source term in MBq t^{-1}	Source Term MBq (unless stated otherwise)	Mechanism and source of production of radionuclide (fission, activation or corrosion product)	Reasoning for assessment of significance (based on half-life, magnitude of source term or discharges, dose impact or as indicator of plant performance)	Significance	BAT Form Required?	Techniques used to minimise activity produced at source	Techniques used to minimise impacts of disposal
Sb-122	2.7 d	Sb-121	1.2	354.72	Corrosion. Isotope is formed by neutron activation of another isotope of Sb. Sb is present in components such as seals and bearings.	Short half-life (<< 1 year), low magnitude in source term.	Low	No	As for Cr-51. In addition, there is minimisation of sources that could give rise to isotopes such as Sb-124, Sb-122 in the coolant by reduction in the use of helicoflex seals in favour of graphite seals and the greater use of rotor stops and bearings without antimony. Reinforced leak tightness requirements for active parts, pH control to minimise corrosion (see section 3.3.1 of Sub-chapter 8.2 of the PCER).	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER).
Sb-124	60.2 d	Sb-123	0.97	286.73	Corrosion. Isotope is formed by neutron activation of another isotope of Sb. Sb is present in components such as seals and bearings.	Relatively short half-life (< 1 year), low magnitude in source term and is not a significant contributor to radiation doses.	Low	No	As for Cr-51. In addition, there is minimisation of sources that could give rise to isotopes such as Sb-124, Sb-122 in the coolant by reduction in the use of helicoflex seals in favour of graphite seals and the greater use of rotor stops and bearings without antimony. Reinforced leak tightness requirements for active parts, pH control to minimise corrosion (see section 3.3.1 of Sub-chapter 8.2 of the PCER).	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER).
Sb-125	2.73 y	Sb-124	11	3251.6	Corrosion. Isotope is formed by neutron activation of another isotope of Sb. Sb is present in components such as seals and bearings.	Relatively long half-life (> 1 year). Low magnitude in source term and is not a significant contributor to radiation doses.	Low	No	As for Cr-51. In addition, there is minimisation of sources that could give rise to isotopes such as Sb-124, Sb-122 in the coolant by reduction in the use of helicoflex seals in favour of graphite seals and the greater use of rotor stops and bearings without antimony. Reinforced leak tightness requirements for active parts, pH control to minimise corrosion (see section 3.3.1 of Sub-chapter 8.2 of the PCER).	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER)
Sr-89	50.5 d		0.3	88.68	Fission. Isotopes of strontium are formed as a result of fission. They are usually confined in the fuel but, in the event of fuel leaks, they can pass into the primary coolant via defects in the fuel cladding. Their presence in the primary coolant is also due to the occurrence of traces of uranium (“tramp” uranium) that can never be completely removed on new fuel assemblies following the manufacturing process. See also Cs-134.	Relatively short half-life (< 1 year). Low magnitude in source term.	Low	No	As for Cs-134	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER). Isotopes of strontium are not thought to be discharged from PWRs (see section 5.1 of Sub-chapter 8.4 of the PCER).

Table 2: Table of radionuclides arising from operation of the EPR (continued)

Radio-nuclide	Half-life	Source	Source term in MBq t^{-1}	Source Term MBq (unless stated otherwise)	Mechanism and source of production of radionuclide (fission, activation or corrosion product)	Reasoning for assessment of significance (based on half-life, magnitude of source term or discharges, dose impact or as indicator of plant performance)	Significance	BAT Form Required?	Techniques used to minimise activity produced at source	Techniques used to minimise impacts of disposal
Sr-90	29.2 y		0.0019	0.561	See Sr-89.	Relatively long half-life (>> 1 year). Low magnitude in source term.	Low	No	As for Cs-134.	Filtration to remove particulate material from liquid and gaseous discharges, use of ion exchange and/or evaporation for removal of dissolved material in liquid discharges (see section 3.3.1 of Sub-chapter 8.2 of the PCER). Isotopes of strontium are not thought to be discharged from PWRs (see section 5.1 of Sub-chapter 8.4 of the PCER).
Xe-133	5.25 d		5000	1,478,000	As for Kr-85.	Relatively short half-life (<< 1 year) and is not a significant contributor to radiation doses but is large in terms of magnitude of source term. Use of carbon-bed delay system to minimise discharges.	Medium	Yes	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
Xe-133m	2.19 d		110	32,516	As for Kr-85.	Relatively short half-life (<< 1 year) and is not a significant contributor to radiation doses but is large in terms of magnitude of source term. Use of carbon-bed delay system to minimise discharges.	Low	Yes (with Xe-133)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
Xe-135	9.09 h		1100	325,160	As for Kr-85.	Short half-life (< 1 d) and is not a significant contributor to radiation doses but is large in terms of magnitude of source term. Use of carbon-bed delay system to minimise discharges.	Low	Yes (with Xe-133)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.
Xe-138	14.2 m		850	251,260	As for Kr-85.	Short half-life (< 1 d) and is not a significant contributor to radiation doses but is large in terms of magnitude of source term. Use of carbon-bed delay system to minimise discharges.	Low	Yes (with Xe-133)	See BAT form for detailed information on techniques used to minimise activity produced at source.	See BAT form for detailed information on techniques used to minimise impacts of disposal.

3. BAT FORMS

3.1 INTRODUCTION

This section presents the BAT forms for the radionuclides that have been identified as significant as discussed in sections 1 and 2, namely carbon-14, tritium, cobalt-60 and cobalt-58, noble gases (krypton and xenon isotopes) and iodine isotopes. The forms contain the following information:

- the mechanism of production of the radionuclide and the process by which it is produced;
- the source term including, where possible, information on the activity before and after the application of techniques to prevent or minimise at source;
- the demonstration that the techniques used to minimise at source are BAT by reference to all other options considered or rejected or by benchmarking against best world practice;
- the demonstration that the techniques used for treating each process stream minimise the overall impact of the disposal of radioactive waste (including evidence of the consideration of disposal via different pathways);
- the demonstration that the techniques used to minimise impacts are BAT by reference to all other options considered and rejected or by benchmarking against best world practice;
- the demonstration that the techniques used conform to the Integrated Waste Strategy (IWS) by cross-referencing to the relevant IWS sections (which is based on the Flamanville 3 Reference Case of the UK EPR GDA design submission); and
- the identification of disposals resulting from the use of techniques to minimise overall impacts and quantification (where possible) of annual volumes and activity of nuclides.

The forms provide information on gaseous, liquid and solid wastes containing each nuclide. Much of the information presented has been drawn from the Pre-Construction Environmental Report (PCER) for the UK EPR but other information has also been used, including international literature relating to waste management and detailed technical reports. The references are provided throughout the forms but are also listed in section 6 of the report.

The BAT form for isotopes of cobalt can also be considered as providing information relevant to the minimisation, management and abatement of other corrosion products listed in Table 2.

3.2 BAT FORM FOR CARBON-14

1. Mechanism of production of the radionuclide and how it relates to the process by which the nuclide is produced	The production of C-14 in the EPR occurs via two main mechanisms, namely the neutron activation of N-14 and O-17 [IAEA Technical Report 421 Management of Waste Containing Tritium and Carbon-14, July 2004]. The two main sources of nitrogen and oxygen which give rise to the C-14 are the coolant and the fuel.

Reaction	Thermal neutron cross-section (barn)	Isotopic abundance (%)
N-14 (n, p) -> C-14	1.82 barn	99.63
O-17 (n, α) -> C-14	0.24 barn	0.04

Another minor mechanism contributing to C-14 is the reaction C-13 (n, γ) -> C-14, which occurs due to the presence of dissolved carbon in the coolant. The neutron capture cross-section of C-13 is relatively low (0.9×10^{-3} barn), and this reduces the magnitude of C-14 generated in comparison with the N-14 and O-17 reactions, by approximately three orders of magnitude.

The formation of C-14 in coolant and fuel is discussed below.

IAEA Technical Report 421 indicates that the production of C-14 depends on the enrichment of the fuel, the relative mass of the fuel and moderator, the concentration of nitrogen impurities in the fuel and structural materials and the temperatures of the fuel and moderator. On this basis precise calculations of C-14 production are difficult to carry out and for the purpose of the demonstration of BAT emphasis is being placed on the review of PWR operating experience feedback with respect to the estimation of C-14 discharges.

BAT FORM FOR CARBON-14 (continued)

	1.1 Coolant Oxygen is abundant in the reactor coolant system (primarily in water and also in the chemicals used for reactivity and chemical control, e.g. boric acid, lithium hydroxide), whereas nitrogen is present in relatively low concentrations. Production estimates (PCER Sub-chapter 6.2, point 1.2.1) indicate that the oxygen reaction is the dominant C-14 production mechanism in the coolant (80% - non variable), whereas the nitrogen reaction is smaller (~20% variable depending on the nitrogen concentration in the primary coolant) which depends upon the reactor operational regime. The EPR uses nitrogen gas to control the pressure in particular in the Volume Control Tank (VCT), the Coolant Storage and Treatment System TEP [CSTS] tanks and the Reactor Boron and Water Make-up System REA [RBWMS] tanks and also uses it as a purge gas. Under these conditions, and upon addition of make-up water, dissolved nitrogen will be present in the coolant, the concentration of which depends upon the partial pressure in the system. **1.2 Fuel** Carbon-14 is formed in large quantities in the fuel from oxygen and the UO_2 and nitrogen impurities. This C-14 is confined within the fuel cladding and removed from the reactor with the fuel. IAEA Technical Report 421 indicates that for light water reactors the zirconium alloys cladding of fuels is the dominant source of C-14 containing 40% – 50% of its total, and is used as the cladding on all LWR fuel. C-14 from fuel is not taken into account in normal reactor discharge arisings and would only be discharged during major fuel cladding failures. Even then it is uncertain how much would enter the coolant, as the transfer process by which the C-14 would migrate from the fuel to the coolant is not clearly understood. It is judged that this uncertainty is acceptable due to the low radiological impact that could occur as a result of such an event. It is assumed in the PCER (Sub-chapter 6.3, section 6.3.1) that the fuel cladding is essentially impermeable with respect to C-14, a view which is supported by information in IAEA Technical Report 421 (section 4.1) which indicates that most of the C-14 in fuel will pass to reprocessing plants. Information on fuel failure rates is not readily obtainable due to commercial confidentiality issues but any problems with fuel cladding would be expected to be indicated by increased discharges of fission products such as Cs-137 and noble gases (e.g. xenon, krypton). The monitoring of C-14 in liquid discharges would also provide an indication of any increased discharges of C-14 into the primary coolant. The monitoring of C-14 in gaseous and liquid discharges is discussed in sections 2.3 and 6 of the PCER Sub-chapter 8.4 Implementation of BAT within monitoring procedures.

BAT FORM FOR CARBON-14 (continued)

1.3 Other Sources

C-14 in gaseous form is also produced in the ‘aeroball’ system, which uses nitrogen gas to propel vanadium marbles/pellets through the reactor (for neutron flux measurements). The assessed annual production is very low (1.5 GBq y^{-1} as discussed in section 7.3.1 of Sub-chapter 6.3 of the PCER).

1.4 Chemical and Physical Form

The chemical environment of a PWR primary system is reducing because the concentrations of dissolved hydrogen are sufficiently high to consume radiolytically generated oxygen species and therefore diminish corrosion effects [Life cycle and management of carbon-14 from nuclear power generation, M-S Yim and F. Caron, Progress in Nuclear Energy 48 (2006) 2 – 36 and references therein]. Large fractions of the C-14 (in PWR stations) rapidly associate with the hydrogen or trace quantities of organic matter present in the reactor coolant, once the C-14 is formed in the coolant. The predicted forms include CO_2 (plus carbonic acid and bicarbonate), and simple organic carbon compounds of mixed oxidation states. Species formed under specific conditions (e.g. reactor environment at high temperature) might, however, not be stable under ambient conditions for example on ion exchange resin columns and during sampling and analysis of coolant.

Experimental investigations have shown that C-14 exists in PWR coolant systems mainly as organic carbons (58–95%) [Vance, J.N., Cline, J.E., Robertson, D.E., 1995. Characterization of Carbon-14 generated by the Nuclear Power industry, TR-105715. EPRI, Palo Alto, California, Electric Power Research Institute, (referenced within M-S Yim and F.Caron, as above)].

International data [IAEA Technical Report 421, section 3.1.3] indicates that the airborne C-14 released from PWRs is predominantly hydrocarbons (75% – 95%), mainly methane, with only a small fraction as $14CO_2$. As a first approximation for the EPR (PCER Sub-chapter 6.2, section 1.2.1) it is assumed that atmospheric discharges are 80% methane ($14CH_4$) and 20% carbon dioxide ($14CO_2$). This is consistent with the published international data for PWRs from IAEA Technical Report 421.

From the reactor operation described in PCER Sub-chapter 6.3, sections 6.3.2 and 7.3.1, the C-14 produced is discharged mainly in liquid or gaseous form, with an estimated ratio of approximately 80% – 95% gaseous and 5 to 20% as solid and liquid. This is based on operating experience feedback of PWRs. IAEA Technical Report 421 indicates that the majority of C-14 released from reactors is contained in airborne effluents and only a small amount is in the form of liquid effluents while M-S Yim and Caron (referenced above) indicates that 95% of C-14 is expected to be released in gaseous form. Further information on the chemical and physical form of C-14 is provided in section 4 of this BAT form.

BAT FORM FOR CARBON-14 (continued)

2. Source term (before application of techniques to prevent or minimise at source have been applied) and activity in reactor following the application of techniques to prevent or minimise the source term	As noted in section 1 of this BAT form, the source term for C-14 is difficult to predict with any certainty because of the many factors which affect its production. This issue is discussed in some detail in sections 6.3.1 and 6.3.2 of Sub-chapter 6.3 of the PCER. Calculations and operating experience feedback have estimated that the performance for gaseous discharges of C-14 from the EPR during normal operation is expected to be around 350 GBq y^{-1}, with a maximal discharge value expected at 700 GBq y^{-1}. The maximum discharge source term for gaseous C-14 in the PCER is 700 GBq y^{-1}. This maximum figure is based on the following factors: • review of operating experience feedback from the KONVOI reactors in Germany. These reactors are considered to be better operational comparators for EPR C-14 discharges than the 1300 MW(e) reactors in France. This is because, like the EPR, they are based on the use of a nitrogen cover-gas for the chemical volume and control system (RCV [CVCS]) as opposed to the hydrogen atmosphere used for the 1300 MW(e) reactors (and for Sizewell B); • this operating experience feedback indicates that annual C-14 discharges from the KONVOI reactors have been very variable but they are generally higher than those from the 1300 MW(e) reactors, because of the greater use of nitrogen as noted above and the increase in power. The highest annual discharge of C-14 recorded in the period between 1995 and 2003 was 700 GBq; • the most conservative estimate of C-14 production in the primary coolant made in the PCER (Sub-chapter 6.3, section 6.3.1) was based on a nitrogen concentration of 52 ppm in the RCP [RCS] (Reactor Coolant System), noting that a reasonable operating concentration is considered to be 10 ppm. This conservative estimate would give rise to an annual production of C-14 of 620 GBq y^{-1}, which is within the range of operational discharges from the KONVOI reactors. Small amounts also arise from the reactor pit atmosphere and the "aeroball" systems used for neutron flux measurement but these sources add only a few GBq to the total. (Note: the figure quoted in the source term in Table 2 of this document of 518 GBq y^{-1} is based on a nitrogen concentration of 27 ppm). The information above is provided to substantiate, as far as possible, the figure quoted (620 GBq y^{-1}) for the source term in the PCER noting the uncertainty as to the magnitude of the source term for C-14. It is important to note that the "source term" is considered to be proportional to both the size of the reactor and the rate of power production and is dependent on the volume of water subject to neutron flux and the nitrogen content of the primary coolant. The production of C-14 has been assessed using the computer code DARWIN, a specific model for the evaluation of activation products in the reactor core. This computer code is recognised by the French safety authority and has been validated by means of a comparison with experimental results. Further work is being carried out to collect information on operating experience feedback relating to C-14 production in PWRs to enable more accurate assessment of discharges.

BAT FORM FOR CARBON-14 (continued)

	Calculations and operating experience feedback have estimated that the performance for the liquid discharges of C-14 from the EPR during normal operation is expected to be around 23 GBq y^{-1}, with a maximal discharge value expected at 95 GBq y^{-1}. The maximal discharge value is an upper estimate which takes account of the use of nitrogen gas in the primary circuit and takes account of uncertainty in the partitioning of C-14 between solid, liquid and gaseous wastes. Taking account of both liquid and gaseous discharges estimates under normal performance approximately 6% of C-14 would be discharged as liquid while for the maximum discharges approximately 12% would be discharged as liquid. These figures are well within the operating experience feedback as documented in IAEA Technical Report 421 (see section 1). The estimates of C-14 production provided in the PCER are based on the reactor design, which incorporates measures to apply BAT to minimise the activity at source. It is not possible to provide an estimate of activity before and after the application of measures to minimise activity at source as this is considered to be inherent in the design.
3. Demonstration that the techniques used to minimise at source are BAT by reference to all other options considered and rejected or by benchmarking against best world practice	The Gaseous Waste Treatment Processing System (TEG [GWPS], see PCER Sub-chapter 6.4) is based on the KONVOI design. It is a semi-closed loop system for treatment of aerated effluent, which differs from the 1300 MW(e) design. This design enables better treatment of peaks in activity when switching to cold shutdown. Its main characteristics are: • sharing of the TEP [CSTS] and REA [RBWMS] tanks cover gas limits the volume of the gaseous waste in normal operation by maintaining a constant gas volume when transferring water; • continuous nitrogen flushing of the tanks cover gas. This avoids the accumulation of H_2, by standardising gas treatment irrespective of whether its composition contains H_2 or O_2; • recombining hydrogen (H2) and oxygen (O2) in the purge gas to water using a catalytic recombiner. This reduces the concentrations of these in the gas that is recirculated in the system to less than 0.3% and 0.1% respectively. This minimises oxygen in liquids in the various tanks to allow these to be reused in the primary circuit and maintains hydrogen in the tanks below an explosive limit of 4% (eliminating a potential internal hazard). The effect of the catalytic recombiner on Carbon-14 speciation has not yet been fully determined: however, the total released activity of C-14 stays unchanged; • recycling of gases. This limits the volume of gaseous waste in normal operation (with no impact on reducing overall C-14 discharges). • providing radioactive decay of noble gases (xenon is kept for at least 40 days and krypton for at least 40 hours) using three charcoal adsorption delay beds (with no impact on reducing overall C-14 discharges).

BAT FORM FOR CARBON-14 (continued)

As noted in section 1 of this BAT form the majority of discharges of C-14 into the environment (80% – 95%) are in the gaseous form, with typically only 5% to 20% being discharged in liquid and solid wastes. The majority of C-14 is degassed during the treatment of the primary effluents in the Coolant Storage and Treatment System (TEP [CSTS]) and directed into the Gaseous Waste Processing System (TEG [GWPS]) to be discharged as gaseous effluent at the NAB stack via the DWN [NABVS]. Gaseous species containing C-14 are neither retained in the filters and delay beds used in the TEG [GWPS] nor in the filters used in the DWN [NABVS].

A proportion of C-14 is contained in the primary liquid effluents which may be retained on filters and ion exchange resins used in the various effluent treatment systems, thereby giving rise to solid wastes. Abatement techniques for discharges are discussed in section 5 of this BAT form. It is anticipated that increased recycling of effluents incorporated into the design of the EPR increases degassing and thus more of the C-14 is transferred into gaseous effluent than would otherwise be the case if increased recycling did not take place. However, this cannot be readily quantified because of the complex behaviour of the gas-liquid system and the many variables which influence the partitioning between gaseous and liquid effluents.

As noted in section 1 of this BAT form, the EPR design differs from those of other 1300 MW(e) units in its use of nitrogen instead of hydrogen in particular in the Volume Control Tank (VCT) as a scavenging cover-gas. This design choice reduces the non-radiological risks associated with the use of hydrogen gas but has the radiological impact of increasing the source term (and thus discharges) of C-14 in comparison with those reactors which do not use nitrogen for this purpose. The additional C-14 introduced as a result of this change has been assessed to be of the order of 10%, based on the information presented in section 6.3.1 of Sub-chapter 6.3 of the PCER, (where the 10% addition is based on a normal operating concentration of 10 ppm for the use of nitrogen). There is some uncertainty on the additional discharges, as for all arisings of C-14, but the use of higher nitrogen concentrations would be expected to result in increased C-14 discharges.

The use of nitrogen is based on evolution of the design of PWRs, taking account of the operating experience feedback of KONVOI reactors in Germany which use nitrogen instead of hydrogen. This change reduces the non-radiological risks associated with the storage of large volumes of hydrogen, thereby enabling a reduction of the volume of stored hydrogen of 1000 m^3 (section 7.3.2 of Sub-chapter 6.3 of the PCER). No formal options assessment has been carried out and its inclusion in the EPR design is based on benchmarking against world practice in PWR design and operations, as part of the EPR environment design review, and review of non-radiological safety issues.

BAT FORM FOR CARBON-14 (continued)

	The use of nitrogen as a cover gas is considered to be the Best Available Technique for the EPR for the flushing of components in the primary circuit, despite the modest addition to C-14 discharges that the change introduces (in comparison with other PWRs). Flushing of the system is necessary to achieve coolant degasification and also limits the hydrogen content in the system and connected components to less than 4% (the Lower Explosive Limit for hydrogen) by volume (as discussed in section 3.4.1 of Sub-chapter 8.2 of the PCER). It may be noted that the oxygen content is also managed and limited to avoid corrosion of the primary circuit. The nitrogen concentration in the coolant is determined on the basis of measurement of temperature and pressure, taking account of the need to control the hydrogen concentration. In terms of other measures to minimise C-14 production at source, the production of C-14 per unit of fuel is expected to be lower than for the existing 1300 MW(e) EDF reactors as a result of improvements to the design of the core (section 6.3.1 of Sub-chapter 6.3). However, the overall higher power availability of the EPR, together with the additional discharges associated with the use of nitrogen gas in the primary circuit, mean that absolute levels of discharges are expected to be higher overall (see above). The performance of the EPR with respect to discharges of C-14 will be consistent with international operating experience feedback in PWR operation. Solid wastes are not discussed in this section as all C-14 arises in either liquid or gaseous form and is only converted into solid form as a result of effluent treatment processes. Effluent treatment processes are discussed in section 5 of this BAT form.
4. Demonstration that the techniques used for treating each process stream minimise the overall impact of the disposal of radioactive waste (including evidence of the consideration of disposal via different pathways)	This section first discusses the techniques used to minimise the impacts of disposal of the radioactive wastes. Information is first presented on the treatment of gaseous, liquid and solid wastes containing C-14, followed by a discussion of the partitioning of C-14 between gaseous, liquid and solid wastes and the radiological impacts of disposal of these wastes. Information on potential treatment methods considered for the abatement of C-14 discharges is presented in section 5 of this BAT form. The OECD issued a report in 2003 [Effluent release options from nuclear installations. Technical Background and regulatory aspects, OECD 2003] that stated that “in the particular cases of tritium and C-14 currently there are no abatement techniques in place to reduce the discharges of tritium and C-14” (noting that while there are techniques in use to abate C-14 from gaseous discharges from fuel reprocessing plants these are not considered to be practicable for a PWR, further information is presented in section 5 of this BAT form). **4.1. C-14 in gaseous discharges** There is no specific treatment process for the removal of C-14 from gaseous effluents in the EPR, which is consistent with international operating experience feedback of PWR operations [OECD report as referenced above]. The impacts of the disposal into the environment are minimised by the use of a dedicated gaseous waste treatment system, including the use of a discharge stack to discharge effluents at height. This ensures that the gaseous effluents emitted are dispersed into the environment in a manner which minimises impacts to receptors.

BAT FORM FOR CARBON-14 (continued)

4.2. C-14 in liquid discharges and solid wastes

There is no specific treatment process aimed at the treatment of C-14 in liquid effluents in the EPR, as in other PWRs. However some C-14 present in non-recyclable effluents is retained on filters, resins and in the concentrates from the treatment of effluent by evaporation. This is discussed in section 5 of this BAT form. The impacts from disposal of liquid discharges are expected to be minimised by the design of the effluent discharge system, including the design and location of the effluent outfall.

Chemical speciation studies analysed by EPRI [EPRI TR-105715 Characterization of Carbon-14 generated by the Nuclear Power industry] showed that C-14 in the primary coolant of PWR stations is predominantly (58% to 95%) in organic form. This is as expected where hydrogen gas has been used for oxygen suppression, as it combines with the generated C-14 and other trace quantities of organic matter present in the reactor coolant water to form hydrocarbon compounds. Very little is known of the behaviour of the organic C-14 species during the clean-up of reactor coolant. However, the chemical speciation of C-14 on primary demineralisation resins from PWR stations also shows predominantly organic forms (72% - 92%). The attachment mode of these organic C-14 species on the resin is unknown, but probably occurs by some type of sorption process rather than by a normal ion exchange mechanism or by microfiltration for colloidal forms.

Organic C-14 species can also become attached to particulates in the reactor coolant water, as indicated by the observation of the presence of solid carbon on filters in the USA [Miller, 2000 within M-S. Yim and F. Caron, see section 1 for full reference]. This is supported by a report from EPRI which indicated relatively high concentrations of C-14 measured on some primary coolant filter cartridges at a number of nuclear power stations. EPRI reported [Technical Basis for Averaging C-14 Filters, Interim Report: Carbon-14 Source Term Analysis For Encapsulated Filter Waste Forms, EPRI, Palo Alto, CA: 2000. 1000849] that spent cartridge filters have recently emerged as a very important radioactive waste stream for C-14. As part of the nuclear industry’s plant dose reduction efforts [NCRP, 1994; Yim and Ocken, 2001, referenced within M-S. Yim and F. Caron, see section 1], sub-micron size filters have been adopted in some nuclear power plant systems. Experience with these sub-micron size cartridge filters [Miller, 2000 within M-S Yim and F. Caron, see section 1] showed that the concentrations of C-14 are higher in these filters than would normally be expected. This would appear to indicate that particulate containing C-14 is present although the evidence is limited to one study.

Anion-exchange resins are known to be relatively efficient at removing C-14 from nuclear plant waste streams when it is in ionic form (carbonate or hydrogen carbonate). However, the EPRI report and other studies have shown that significant quantities of C-14 remain in reactor coolant after clean-up, predominantly in the organic form.

BAT FORM FOR CARBON-14 (continued)

A study was undertaken [Reference EDEECH060283 – Study on transfer of carbon-14 from primary circuit to liquid effluents. Summary of PAL215 tests] on the second PWR unit at Paluel Power Station to examine the levels of C-14 at various locations in the primary coolant circuit, its treatment system and in effluents when the coolant was not recycled. This included sampling the RCP [RCS], RCV [CVCS], PTR [FPCS] and TEP [CSTS] systems during operations pre-shutdown, in order to assess the activity of C-14 in the RCP [RCS] and PTR [FPCS] systems and to determine the RCV [CVCS] and TEP [CSTS] system's treatment efficiency with respect to C-14. The TEP [CSTS] system was also sampled during shutdown, to assess the influence of a change in the redox conditions on the treatment system for removing C-14. The RCP [RCS], PTR [FPCS], TEU [LWPS] and 0KER [LRMDS] systems were sampled in the later stages of reactor restart, to examine the impact of the shutdown period on the C-14 activity levels present in the primary circuit, the PTR [FPCS] tank and the fuel pool (FB). The samples were used to assess the efficiency of the TEU [LWPS] system as to C-14 generated in the effluents produced during shutdown.

The measurements on the samples taken during operation and shutdown of the reactor showed that the redox conditions play a dominant role in the distribution of the C-14 physico-chemical forms and of the relative efficiency of each treatment method, in particular those in the TEP [CSTS] system. During normal operations, the C-14 activity measurements were practically identical downstream of the RCV [CVCS] purification, the TEP [CSTS] filters and the TEP [CSTS] demineralisers. Also, the C-14 activity levels present in the TEP [CSTS] distillates were of the same order as the activity entering into the system (4800 Bq l^{-1}). These results demonstrate that a major part of the C-14 is degassed in the RCV [CVCS] tank gas blanket and that the physico-chemical forms of C-14 left in the solution were not retained on either TEP [CSTS] filters or resins.

Different results were obtained during the shutdown period compared to normal operations, it was shown that:

- C-14 was mainly found in the form of CO_2 and carbonates;
- the TEP [CSTS] degasser allowed for a significant percentage of the C-14 to be degassed;
- the TEP [CSTS] evaporator allowed for retention of C-14 in the concentrates, however, the C-14 activity measured in the distillates still remained significant (165 Bq l^{-1}) and contributed to the C-14 activity discharged to the 0KER [LRMDS] via the TEU [LWPS].

The results of the sampling campaign undertaken in the study identified the main contributors of C-14 to the liquid effluents from the reactor as:

- the TEP [CSTS] distillates, when not recycled, even if the different TEP [CSTS] purification systems diminish the activity in the liquid phase to a large extent;

BAT FORM FOR CARBON-14 (continued)

- the TEU [LWPS] process drains, as the TEU [LWPS] resins mainly retain the carbonate form of C-14 and hence the demineralisers' downstream activity is still significant.

From the data obtained so far by EDF and from the EPRI work, it is concluded that the reactor coolant treatment and liquid effluent treatment systems do remove some C-14 from liquid effluent but that much of the C-14 remains in the effluent. It is not possible to provide detailed quantification on this beyond the report on Paluel NPP described here and the data reported in the literature references provided. Further studies are ongoing at other French stations to confirm the results obtained at Paluel and build up the operating experience feedback necessary to determine the need or otherwise for further action with respect to C-14 in liquid effluents.

On the basis of the limited information available, which is discussed above, it appears that it is not possible to fully control the partitioning of C-14 between the solid, liquid and gaseous phases, other than maximising the amount in gaseous effluent by means of maximising the efficiency of the degasification process for the primary circuit. This is considered to be the Best Available Technique to minimise radiological impacts of C-14 discharges. There is some partitioning between liquid and solid wastes but the complex redox chemistry associated with C-14 is poorly understood due to its complexity. The liquid effluent treatment systems are optimised for control of coolant chemistry for the purposes of safe reactor operation and minimisation of worker radiation doses.

At present there are limited data on the measurement of C-14 in solid wastes arising from French PWRs because specific information on C-14 is not required for the disposal of solid (which is based on total activity measurements). However, liquid and gaseous discharges are measured for C-14, as discussed in detail in Sub-chapter 8.4 of the PCER. The information in section 1 of this BAT form indicates that up to 15% – 20% of C-14 is present in solid wastes from PWRs. Literature information indicates that for solid wastes from LWRs in the USA approximately half of the C-14 present in solid waste consigned to the Barnwell site from LWRs is present in spent ion exchange resins [M-S.Yim and F. Caron and references therein, see section 1].

It is recognised that the C-14 concentration in solid wastes, in particular ion-exchange resins and cartridge filters, can cause problems with their management, particularly in the UK, and that the limits set for waste repositories are restrictive (section 8 of IAEA Technical Report 421 provides some information on waste acceptance requirements and storage and disposal options). Wastes that could otherwise be managed as Low Level Waste (LLW) may be restricted from disposal to the UK Low Level Waste Repository (LLWR) because of their C-14 content and therefore have to be disposed of as Intermediate Level Waste (ILW). C-14 is a key nuclide in the LLWR's Post Closure Safety Case and is carefully limited in the LLW consigned for disposal to mitigate the fact that it has a high potential mobility and in theory could diffuse into groundwater. The timescale of such a release from the repository would be short in comparison to the half-life of C-14 (5730 years).

BAT FORM FOR CARBON-14 (continued)

For these reasons, concentrating C-14 in solid waste is not considered as BAT. Indeed this issue is raised in Chapter 8 of the UK Strategy for Radioactive Discharges [UK Strategy for Radioactive Discharges, 2006 - 2030, July 2009]. The issue of the partitioning between liquid and solid wastes is discussed further in section 5.

Discussion of the minimisation of the radiological impacts of the disposals of C-14

In order to consider the minimisation of impacts of disposal it is necessary to consider the radiological impacts of the disposal of C-14 in terms of radiation doses from the EPR. These are discussed in detail in PCER Sub-chapter 6.3 and Chapter 11. The methodology for the detailed radiological assessment was based on the use of PC CREAM 98 software tool and is described in Sub-chapter 11.1, section 1.3 of the PCER.

The maximum annual gaseous (900 GBq) and liquid discharges (95 GBq) for an EPR unit were used for the dose impact assessment. However, it should be noted that the operating experience feedback data lead to a more realistic value of 700 GBq y^{-1} as the maximum annual discharge for gaseous C-14. Operation of EPR with this reduced release limit will be possible without operational restrictions. Final calculation of dose to the population will be presented during site specific licensing. Since the maximum release of gaseous C-14 is now assessed to be only 700 GBq y^{-1} instead of 900 GBq y^{-1}, the calculated collective doses will be reduced to about 80% of the values given in the Table below. The table below sets out the calculated doses to the maximally exposed individuals from both liquid and gaseous discharges.

Doses to maximally exposed individuals from Annual Releases of C-14 in gaseous and liquid effluents

Maximally exposed individuals	Exposure from gaseous discharges (900 GBq) (µSv y^{-1})	Exposure from liquid discharges (95 GBq) (µSv y^{-1})	Local resident exposure from both liquid and gaseous discharges (µSv y^{-1})
Adult	3.7	14	17.7
Child	4.1	4.2	8.3
Infant	7.2	1.4	8.6

BAT FORM FOR CARBON-14 (continued)

	It is theoretically possible for a family (adults, children and infants) living in the nearest habitation to be exposed to both atmospheric discharges and to liquid discharges in the marine environment. To determine the dose to these local residents it is necessary to sum the doses from aerial discharges and liquid discharges, and these are shown in the above table for completeness. The highest total dose calculated to a local resident (adult) from exposure to all radionuclides discharged from an EPR is 21 µSv y^{-1}. The contribution of C-14 to this dose is 84% of the dose from discharges. The overall radiation doses from the discharges of an EPR power station are well within the dose limit for members of public (1000 µSv y^{-1}) and the UK government's dose constraint for a single source (300 µSv y^{-1}) [The Radioactive Substances (Basic Safety Standards) (England and Wales) Direction 2000]. It has been estimated [UNSCEAR, 2000] that a natural production rate of 1 petabecquerel (PBq, 10^{15} becquerels) y^{-1} would lead to an individual effective dose rate of 12 µSv y^{-1}. <u>Collective Dose from Annual Releases of C-14 in gaseous and liquid effluents</u> The collective dose is a measure of the radiation exposure in a population and is the sum of effective doses from a given practice or situation to all affected individuals, now and in the future. The environment agencies guidance on prospective doses assessment recognises 500 years as an appropriate measure for collective doses assessments. Collective doses to the populations of UK, Europe and the World, truncated at 500 years, are estimated and presented in PCER Sub-chapter 11.1 section 4. As the site location is an important factor for estimating the dose to the UK population, it is necessary to make some assumptions about the power station location. Among the potential sites where a new EPR reactor could be located, the site which presents the maximum collective dose is presented in PCER Sub-chapter 11.1 section 4.2. The estimated collective doses are presented in the table below.

BAT FORM FOR CARBON-14 (continued)

Collective Dose from Annual Releases of Carbon-14 in gaseous and liquid effluents

		UK	Europe	World
Collective Dose due to C-14 (Man Sv)	**Atmospheric discharges**	0.29	2.31	15.8
	Liquid discharges	0.02	0.15	1.1
	Total	0.31	2.46	16.9
Total Collective Dose (Man Sv)		0.31	2.46	16.9

The results in the table above show that atmospheric discharges are the major contributors to collective dose. The collective dose from atmospheric and liquid discharges from the EPR is dominated by releases of C-14, accounting for almost 100% of the dose for all populations considered.

The collective dose to the UK (0.31 man Sv), European (2.46 man Sv) and World (16.9 man Sv) populations is much lower than the collective dose from C-14 from natural sources to the UK population, estimated to be approximately 480 man Sv, and the world population (~10 billion) of 120,000 man Sv, [European Commission, Guidance on the calculation, presentation, and use of collective doses for routine discharges, Radiation Protection 144, 2007]. The discharges from the EPR make a tiny contribution (0.06%) to collective doses associated with C-14 in the UK.

Potential Short Term Doses

Potential short-term doses have been assessed for gaseous discharges. For the radiological impact assessment of gaseous discharges, it is normally assumed that discharges occur continuously and uniformly over a year. However, during normal operations, short-term gaseous discharges can occur during, for example, outages and start-up or when purging the cooling system. It is possible that such short-term discharges may lead to doses that are higher (or indeed, lower) than would be expected if it were assumed that the discharges are continuous over a year.

BAT FORM FOR CARBON-14 (continued)

The methodology used and calculated doses are presented in PCER Sub-chapter 11.1 section 2, using the release figures for C-14 presented in section 1 of this form. The table below shows the dose to adults, children and infants from all pathways and radionuclides as a result of a single short term discharge. Despite the conservative assumption used, namely that all radionuclides are discharged together in the same release; the total short-term dose is less than the total continuous release dose. C-14 is the dominant radionuclide for these exposure groups and contributes 76%, 85% and 90% of the total dose for adults, children and infants respectively.

Potential Short Term Doses

Radionuclide Group	**Effective Dose per discharge (µSv) for a 24 hour scenario**		
	Adult	**Child**	**Infant**
C-14 short-term dose	0.659	0.758	1.32
Total short-term dose	0.864	0.889	1.46

A number of key points can be drawn from the summary of radiological impacts of the discharges of C-14 and the partitioning of C-14 between gaseous, liquid and solid wastes presented above.

- The highest doses to maximally exposed individuals associated with C-14 (which is the major contributor to doses) are associated with liquid discharges. Given that the dose assessment was based on liquid discharges of 95 GBq and gaseous discharges of 900 GBq it can be inferred that if more of the C-14 was distributed into liquid discharges instead of gaseous discharges then doses would be higher. The dose (µSv) per GBq of C-14 discharged can be calculated to be 0.008 for gaseous discharges and 0.15 for liquid discharges, which means that the dose per GBq from liquid discharges is almost 19 times higher than from gaseous discharges. In terms of doses to maximally exposed individuals the radiological impacts of C-14 disposals are minimised by discharging to the environment in the form of gaseous discharges as opposed to liquid discharges.

- Short term doses associated with C-14 discharges to atmosphere are assessed to be very low.

BAT FORM FOR CARBON-14 (continued)

	• Although collective doses are dominated by gaseous discharges the overall impacts of discharges of C-14 are very low and would not justify a change from gaseous to liquid discharges, particularly in the light of the impact of liquid discharges on doses to maximally exposed individuals. Quantitative information on the C-14 content of solid waste is limited but the information in section 1 of this form indicates that up to 15% – 20% of C-14 discharged from the EPR will be in the form of solid waste. On the basis of the information discussed above on the behaviour of C-14 it is reasonable to assume that most of the C-14 in solid waste is removed from liquid effluent rather than gaseous (ion exchange resins and liquid cartridge filters). On this basis it can be assumed that the partitioning of C-14 on to solid waste from liquid effluent, such as occurs, minimises the impacts of the disposal of liquid effluents. A key question is whether the environmental impacts could be reduced by the application of treatment techniques which remove C-14 from gaseous and liquid effluents and transfer it into solid waste, and whether these could be justified as BAT. This is discussed in section 5 below.
5. Demonstration that the techniques used to minimise impacts are BAT by reference to all other options considered and rejected or by benchmarking against best world practice	A number of techniques [ENTECH080059 EPR – State of the art for abatement of carbon-14 in gaseous effluents] have been considered for the abatement of gaseous and liquid discharges of C-14, which are presented below. <u>Techniques considered for the abatement of gaseous discharges of C-14</u>. **Delay Beds:** Delay beds which are installed on the EPR for the abatement of noble gases will have only a negligible impact on gaseous discharges of C-14 because of its long half-life. The use of the delay beds may result in the presence of some C-14 on the delay beds but the majority of C-14 is discharged to atmosphere. The delay beds have a design life of 60 years. Delay beds are not considered to be feasible for the abatement of C-14 from gaseous discharges. **Alkaline scrubbing:** In this method off-gas would need to be oxidised to ensure all C-14 is present as $14CO_2$ and then scrubbed to remove $14CO_2$. The scrubbing medium may be a liquid (typically sodium hydroxide solution), a slurry (typically lime) or a solid (typically lime or barium hydroxide). When a liquid medium is used, the spent liquid is treated to precipitate the C-14 as a solid form (typically using lime or barium hydroxide). Liquid scrubbing is used in nuclear fuel reprocessing plants. For example THORP at Sellafield captures gaseous C-14 (from the fuel dissolution unit) using simple sodium hydroxide absorption columns. This precipitates the C-14 in an insoluble form, e.g. barium carbonate, which can eventually be solidified. In the Hague reprocessing plant, C-14 is captured in the liquid form before discharge. Its annual environmental report (2007) shows that C-14 is partly absorbed in a sodium hydroxide solution which is neutralised using nitric acid, and then diluted in tritiated water prior to discharge.

BAT FORM FOR CARBON-14 (continued)

This does not constitute a realistic capture method, but is a method for the transfer of CO_2 from gaseous to the liquid phase. As noted in section 4 it is shown that gaseous discharges have a lower radiological impact per unit discharge than liquid discharges for the EPR. [Rapport environnemental, social et sociétal 2007, Etablissement de La Hague", AREVA NC].

A prototype lime slurry scrubber has been developed for the treatment of CANDU reactor off-gas. The results indicated that the slurry scrubber could remove low concentrations of CO_2 (0.04 – 1000 ppmv of $14CO_2$ + $12CO_2$) from gas streams with a high single pass removal efficiency, high reagent utilisation, relatively low system pressure drop and phase entrainment. However, the system has not yet been used on an operational reactor.

Work on dry barium hydroxide scrubbers for treating PWR and BWR off-gas has been performed in the US. This demonstrated that the process is capable of high CO_2 removal efficiencies and high reactant utilisation, and has acceptable operational characteristics at near ambient conditions. There is no information available however to suggested that the process has been taken forward and fitted to an operational reactor.

The removal of C-14 from gaseous streams is discussed in IAEA Technical Report 421 (see section 1 for full reference). It notes that most of the available techniques, such as alkaline scrubbing, remove C-14 as CO_2. The main disadvantage of these methods is that a large amount of secondary waste is generated, particularly if the waste stream contains large amounts of $12CO_2$. Although no detailed options assessment has been carried out for the EPR, a number of issues would need to be resolved before such a technique could be considered for application on an operational reactor, including the techniques and equipment needed to oxidise methane and other organic components to carbon dioxide, the handling of the secondary wastes, the subsequent conditioning and packaging of the wastes to meet the CfA for the disposal routes (ILW, LLW or VLLW), radiation doses to workers and other safety issues associated with such treatment. The cost effectiveness of the application of these techniques in relation to the radiological impacts of gaseous discharges remains to be demonstrated.

As discussed above, a number of alkaline scrubbing processes, such as those on fuel reprocessing plants, have been investigated for use on PWRs but none have been fitted on operational reactors. On this basis the use of alkaline scrubbing is not considered to be world best practice and is not given further consideration.

Fluorocarbon or ethanolamine absorption: In this method off-gas is passed through a cooled fluorocarbon solvent. $14CO_2$ is absorbed and subsequent stripping yields a stream of concentrated $14CO_2$. This needs to be preceded by an oxidation stage to ensure all C-14 is present as CO_2. The method is intended principally for treating reprocessing plant dissolver off-gas. Pilot-scale facilities have been operated in US and Germany.

BAT FORM FOR CARBON-14 (continued)

	In the report [The Techniques of the Engineer, G1 815 Carbon Dioxide, P. Le Clourec, 2008] gas scrubbing with ethanolamine ($HOCH_2CH_2NH_2$) is one method identified for the removal of CO_2 from gaseous streams used in the chemical industry. It is also discussed in IAEA Technical Report 421. This method involves absorbing the CO_2 into an ethanolamine solution at ambient temperatures in a scrubbing tower; the solute is then steam stripped back out of the scrubbing solution in a second tower. Such a process removes CO_2 from one gas stream and produces another gas stream richer in CO_2. A reduction factor of 2 to 7 is claimed in the gaseous discharged stream when treating initial concentrations of 3% to 15% CO_2. A particular operating problem when using ethanolamine is its oxidation to corrosive oxalic acid and glycine, which has been experienced in the gas industry. This could possibly be amplified in a radiochemical application. In addition to either of these absorption techniques a product solidification technique would be required if C-14 was to be prevented from being discharged to the environment. The development of this waste management option in the nuclear industry has been focussed on providing a treatment solution for discharges from reprocessing, which potentially involves much higher activity levels of Kr-85 and C-14. The suitability of this method for the lower concentration of carbon dioxide found in gaseous waste streams in the EPR (< 0.1%) and other PWRs is not demonstrated and, as with alkaline scrubbing, a number of issues would need to be resolved before the technique could be considered for use on an operational reactor. This method has not been adopted on any operational reactor and, as for alkaline scrubbing, cannot be considered to be world best practice and is thus not given further consideration. **Molecular sieve adsorption**: For this method off-gas needs to be oxidised to ensure all C-14 is present as $14CO_2$. Oxidised off-gas is passed through a molecular sieve, which captures the $14CO_2$. Concentrated $14CO_2$ is subsequently desorbed from the molecular sieve and captured in an alkaline scrubber (liquid, slurry or dry, see above). Molecular sieves can be regenerated in service via electrodesorption, over many cycles. This method can only be used with low concentrations of CO_2 (< 1% to 2%), which correspond with the level of < 0.1% CO_2 in EPR waste gaseous discharge. However, this method is reported to be slow and not suited to industrial scale use and is so far only used for sampling of low concentration outputs in the stack. Moreover, there still remains the problem of managing the trapped C-14 (in the gaseous CO_2). It could be stored as solid waste in the saturated molecular sieves, or the sieves can be regenerated but this still would leave the issue of managing a concentrated gaseous CO_2 waste stream that would have to be captured in an alkaline scrubber (liquid, slurry or dry).

BAT FORM FOR CARBON-14 (continued)

For efficient and reliable use of molecular sieves a constant environment is required, i.e. stable temperatures and stable pressures. This is not the case for the EPR gaseous waste stream, where flow rates can vary from 0.2 m^3 h^{-1} (stable running) up to 100 m^3 h^{-1}, i.e. a factor of 500.

Some development work has been undertaken on this treatment option in the nuclear industry [IAEA Technical report 421, July 2004] but no pilot or prototype installations have been constructed. This management option is not proven on a PWR, cannot be considered to be world best practice and is not given further consideration.

Cryogenic distillation: Noble gases, water vapour and carbon dioxide can be frozen out of a gaseous waste stream. Then, fractional distillation at liquid nitrogen temperatures produces a concentrated stream of carbon dioxide (including 14CO$_2$), which can be removed from the waste.

According to the “The Techniques of the Engineer, G1815” this technique has only been used on small scale demonstration installations, working on the solidification of CO$_2$ at a temperature of -140°C and at atmospheric pressures, using a refrigerated heat exchanger system. This option has not been developed sufficiently to be a credible option for application on an operational reactor and is therefore not given further consideration.

Gas hydrates: "The Techniques of the Engineer, G1815" describes this process as being at the R&D phase, with a single notable use, on offshore gas and oil platforms, which release very high CO$_2$ concentrations. It is not considered suitable for use for applications where the CO$_2$ concentration is low, such as for the EPR gaseous waste stream, and is not given further consideration on this basis.

Other methods: Several other methods of separating and fixing C-14 from nuclear plant’s operational releases of 14CO$_2$ are being investigated. These include decomposition of CO$_2$ using microwave discharges [Journal of Nuclear Science Technology 37 (9)] and isotope separation of CO$_2$ using plasma chemical reactions [Journal of Nuclear Science Technology 38 (109)]. Once separated, it is envisaged that the elemental C-14 would be stored as elemental carbon. However, these techniques are currently in the early stages of development and have not been demonstrated at an industrial scale or used on a PWR. They are thus not given further consideration.

Review of the potential techniques for the removal of C-14 from gaseous effluents indicate that while there are a number of potential methods for removal of carbon from gaseous waste streams, none have been identified as being used on operational reactors and are thus not world best practice while others are not at a sufficiently advanced state of development to enable their use on an operational reactor.

BAT FORM FOR CARBON-14 (continued)

	The methods identified are either: • proven on an industrial scale for non-nuclear applications but are not proven for use on the EPR gaseous waste stream. They are used on waste streams containing the much higher concentration of CO_2 in combustion smoke (3% to 15% CO_2 composition) rather than < 0.1% CO_2 as in the case of the EPR waste stream; • used for treating effluent arising from nuclear plants with a higher concentration of C-14, as in gaseous waste streams at fuel reprocessing plants; • or the methods are in the early stages of development and have not been either proven on an industrial scale or fitted to an operational PWR. IAEA Technical report 421 concluded that methods for the separation of C-14 from gaseous wastes are costly and require high energy consumption and that application of these separation technologies may therefore be limited by their high cost. On the basis of this conclusion, and the fact that the radiological impacts of disposal of gaseous effluents without treatment are low in comparison with limits and constraints, it is concluded that the method used for minimising the impacts of the disposal of gaseous effluents on the EPR is the Best Available Technique. <u>Techniques considered for the abatement of liquid discharges of C-14</u> As noted in section 1 of this BAT form, it is estimated that only a small proportion of the C-14 initially present in the liquid phase is discharged in liquid effluents. It has been assumed that between 80% and 95% of the C-14 produced in PWRs is released in gaseous effluents with the remainder 5% to 20% in liquid and solid wastes. The uncertainties are associated with the chemical form of C-14 which is a determining factor of C-14 behaviour in the processes of the plant and are discussed in section 4 of this form. Section 6.3.2 of Sub-chapter 6.3 of the PCER, in particular Figure 10, provides information on the production and discharge of C-14 in liquid discharges. The operation of the demineralisers in the RCV [CVCS] will result in some removal of C-14 from the liquid (and thus conversion into solid form on the ion exchange resins) in the primary circuit and thus reduces the amount in the non-recycled distillates resulting from evaporation of primary coolant. However, the effectiveness of the demineralisers in the removal of C-14 cannot be predicted with confidence in the absence of definitive information on the proportion of C-14 present in inorganic and organic forms, as discussed in section 4 of this BAT form. Some removal will also occur as a result of filtration (see section 4). The use of the RCV [CVCS] and the recycling of primary coolant are optimised for either recycling of materials (e.g. boron) for reactivity control or removal of materials for corrosion control (e.g. Co-60) to support safe reactor operation.

BAT FORM FOR CARBON-14 (continued)

	The non-recycled distillates arising from the evaporation of primary effluent and the effluents arising from the Liquid Waste Processing System TEU [LWPS] contain the residual C-14 in liquid discharges which is discharged to the environment. Further treatment, such as additional demineralisation, could in theory reduce the C-14 content further but would result in the production of secondary wastes, the disposal of which may be problematic (as discussed in section 4). Further demineralisation would only be effective largely for removal of C-14 in inorganic form and it is not clear what further abatement of C-14 would be achieved, given that the non-recycled distillates have already been demineralised, as discussed above. The volume of distillates generated per annum has been estimated to be approximately 1700 m^3 [ECEF050113 EPR Flamanville – Design study and validation of the main characteristics of the liquid waste processing system – TEU - March 2006, EDF]. The total volume of liquid effluent discharged per year from Flamanville units 1 and 2 is of the order of 19,000 m^3. If no segregation is performed, the evaporation of such volumes would require significant amounts of additional energy whilst conversion to solid waste would produce large volumes of solid waste. Estimating the energy necessary to evaporate the liquid effluents at 2500 kJ kg^{-1}, it would require approximately 13 GWh to evaporate the 19,000 m^3 of liquid effluents produced per year by Flamanville units 1 and 2. The practicability of further reduction of liquid discharges by means of either additional evaporation or solidification of the distillates and residual effluents from the TEU [LWPS] has not been established but there are on-going programs on C-14. However, the proposed practices for the management of liquid effluent for the EPR are consistent with international best practice for PWRs and take account of operating experience feedback. It is noted that IAEA Technical Report 421 (section 5.2 Removal of C-14 from liquid waste) states that “the amount of C-14 in liquid waste produced during reactor operations is relatively small in comparison with solid and gaseous wastes. The active liquid waste treatment system installed in a reactor is designed for the removal of other radiochemicals (e.g. Co-60 and Cs-137) and some toxic chemicals. C-14 in liquid water is expected to be removed by this treatment system. Other than solidification of carbonates arising from off-gas scrubbing in fuel reprocessing plants, no other techniques for the removal of C-14 from liquid wastes have been identified”. On the basis of the above discussion it is concluded that impacts of the presence of C-14 in liquid effluents have been minimised as a result of the treatment processes applied and by the discharge of the majority of C-14 in gaseous form. Although the doses from liquid discharges are higher than for gaseous discharge they are nonetheless low in comparison with dose limits and constraints. The methods used for removal of C-14 from liquid effluents are consistent with international best practice and are considered as Best Available Techniques.

BAT FORM FOR CARBON-14 (continued)

<table>
<tr><td>6. Demonstration that the techniques used conform to the IWS (cross-reference to IWS sections)</td><td>The management of wastes are described in the following sections of the Integrated Waste Strategy (IWS) document:
Section 5 Integrated Waste Management Strategy
Section 5.2 Operational radioactive wastes
Section 5.2.1 Solid wastes
Section 5.2.2 Liquid radioactive wastes
Section 5.2.3 Gaseous radioactive wastes.
It may be noted that the IWS does not provide detailed discussion on C-14, which is discussed in this form.</td></tr>
<tr><td>7. Identification of disposals resulting from the use of techniques to minimise overall impacts and quantification of annual volumes and activity of nuclides</td><td>The table below provides information on the expected and maximum values of discharges of C-14 in liquid and gaseous discharges for the UK EPR.
<table>
<tr><th></th><th>Expected annual performance (excluding contingency) (GBq)</th><th>Maximum annual radioactive discharges (GBq)</th></tr>
<tr><td>Carbon-14 in liquid discharges</td><td>23</td><td>95</td></tr>
<tr><td>Carbon-14 in gaseous discharges</td><td>350</td><td>700</td></tr>
</table>
With regard to operational VLLW and LLW, LLWR technical experts have performed their review of the information provided on the D1 forms and C-14. The operators of LLWR have provided a letter confirming the acceptability, in principle the acceptability, of the UK EPR LLW waste streams for treatment and disposal to LLWR [Form D1 Application: UK EPR Project, from LLW Repository Ltd. 320.L.027. December 2008]. The assessment of the disposability of operational ILW is still in progress, but no issues associated with the disposal of C-14 are anticipated at present.</td></tr>
</table>

3.3 BAT FORM FOR TRITIUM

1. Mechanism of production of the radionuclide and how it relates to the process by which the nuclide is produced.	**1.1 Tritium Production in the Primary Coolant** There are three main sources of tritium in the primary coolant: • boron; • lithium; • deuterium. Production of tritium in the primary coolant is a combination of the following reactions: • B-10 (n, 2α) H-3; • B-10 (n, α) Li-7 (n, αn) H-3; • Li-6 (n, α) H-3; • Li-7 (n, αn) H-3; • H-2 (n, γ) H-3. **The Production from Boron in the Primary Coolant** Boron is used in the primary coolant as a mean of controlling the reactivity of the reactor. It is injected in the form of boric acid. Natural boron contains approximately 20% of B-10, the residual being B-11. It is the B-10 which is important because it has a very high thermal neutron absorption cross-section (3837 barns) whereas the cross-section for B-11 is only 5 millibarns. Boron is added to the coolant in the form of boric acid, which is readily soluble and spreads uniformly throughout the circuits. The concentration of boric acid in the primary circuit depends on the way the reactivity is managed.

BAT FORM FOR TRITIUM (continued)

The Production from Lithium in the Primary Coolant

Lithium is injected into the reactor coolant to increase the pH of the coolant and off-set the effect of the boric acid. It is injected in the form of lithium hydroxide. Natural lithium contains approximately 7.5% of Li-6, the remainder being Li-7. It is the Li-6 which is most significant to tritium production because it has a very high thermal neutron absorption cross-section (953 barns) whereas the cross-section for Li-7 is only 37 millibarns.

The Production from Deuterium in the Primary Coolant

Deuterium is a natural isotope of hydrogen. It has a low capture cross-section for the production of tritium of 0.5 millibarns and as such is a lesser contributor to the production of tritium for the EPR.

1.2 Tritium Production from Indirect Sources

Potential indirect sources of tritium are:

- the fuel which contains tritium produced from ternary fission reactions;
- He-3 (n, p) H-3 from the helium pressurising the fuel rods;
- The beryllium in the secondary neutron sources.

Tritium from the Fuel

The fuel is a large potential source of tritium. However, the evidence is that only a very small proportion of this, if any at all, is released through the fuel clad (PCER Sub-chapter 6.3 section 6.2.1).

Helium pressurising the Fuel Rods

This source is not significant compared to ternary fission. In addition, the source is contained within the fuel pin and only a very small proportion is released into the coolant through the Zircaloy cladding.

BAT FORM FOR TRITIUM (continued)

	Secondary Neutron Sources The secondary sources are aimed at demonstrating the availability of the source neutron channels by ensuring a count rate above background. They consist of antimony and beryllium in a sealed canister of stainless steel. However, the use of beryllium to generate the neutrons also produces tritium in increasing amounts as the neutron source canisters are irradiated. The production of tritium is due to the following reactions: • Be-9 (n, α) Li-6; leading to: • Li-6 (n, α) H-3. All tritium produced in the neutron sources is released to the coolant since the stainless steel containment is permeable to tritium, consistent with the large releases of tritium seen from the UK Advanced Gas Cooled Reactors in which the fuel is also clad in stainless steel. **1.3 Chemical Form** Because of the way it is produced, most tritium resulting from the operation of a PWR is present in the liquid form. As such, the majority of tritium will be discharged in the liquid effluents (> 95%). Once formed, tritium is present in the various tanks and storage ponds as tritiated water. The main source of gaseous tritium is from the evaporation of pools containing tritiated water, in particular the IRWST pool (except the part of the tritium recondensing on the cold parts of the EVR [CCVS]), as well as from the evaporation of pools in the Reactor and Fuel Buildings.
2. i. Source term (before application of techniques to prevent or minimise at source have been applied) and; ii. Activity in reactor following the application of techniques to prevent or minimise the source term.	**2.1 Source Term** The production of tritium is an almost direct function of the energy produced, excluding the significant contribution of the secondary source rods. The activities given in section 2.2 are after optimisation of the source term. They incorporate the use of gadolinium to reduce B-10 initial inventory, enrichment of natural boron and the use of Zircaloy clad for fuel. It is difficult to quantify the reduction in activity achieved by minimisation at source. However, an unabated tritium source term might be expected to result in discharges similar to those expected from use of a facility in which fuel is clad with steel and not Zircaloy (unless it is argued that Zircaloy is not abatement but standard practice), the boron used is not enriched and no gadolinium is used (See section 3.1 of this form for discussion of the use of boron and gadolinium).

BAT FORM FOR TRITIUM (continued)

2.2 Activity in Reactor Following Optimisation of the Source Term

The actual concentration of tritium in the primary coolant depends on the liquid effluent treatment policy (recycling or release) and the initial boron concentration in the primary coolant. Based on average values measured for the fleet of N4 units, the tritium concentration in the primary system is estimated at 37 TBq t^{-1} (PCER Sub-chapter 6.1 section 2.3).

Based on section 6.2.1 of Sub-chapter 6.3 of the PCER, the source contributions to the maximum estimated liquid discharges of tritium are presented in the table below.

Maximum Annual EPR Production of Tritium

Source of Tritium	Arisings in a 22 Month Cycle with High Lithium Chemistry (TBq) (6 ppm Li, 99.90% Li-7)
Boron	~ 45
Lithium	~ 18
Deuterium	~ 3
Secondary Source Rods	~ 9
Total	75

BAT FORM FOR TRITIUM (continued)

	The maximum levels of tritium production from lithium are based on a high lithium chemistry with initial concentrations of 6 ppm and tuned to the boron concentration to maintain a constant pH at 300°C. The relationship between boron and lithium concentrations is discussed in section 3. Recent work undertaken by British Energy has considered a comparison of tritium production levels by Sizewell B taking into account the increased reactor power and higher lithium concentration, which shows a close correlation with those predicted for the EPR. The fuel is a large potential source of tritium. Estimated production in the GDA is approximately 1000 TBq y^{-1} for a 1450 MW(e) PWR. However, the GDA has an estimated release fraction of 0.01%, equivalent to 0.1 TBq y^{-1} (PCER Sub-chapter 6.3, section 6.2.1). This means that under normal operation, fuel is a negligible contributor of tritium to discharges from an EPR.
3. Demonstration that the techniques used to minimise at source are BAT by reference to all other options considered and rejected or by benchmarking against best world practice.	The following measures have been incorporated into the EPR design to minimise the production of tritium. **3.1 Burnable Poison and Boron Concentration** It is possible, by enrichment, to increase the level of B-10 in the boron in boric acid. This would allow a reduction of the level of boron in the primary coolant to maintain the same B-10 concentration in core for reactivity control. Reducing the bulk boron level could have an effect (also a reduction) on the level of lithium required in coolant for chemistry control purposes. With respect to production of tritium from boron, however, since B-10 is the parent isotope and its concentration in core is unaltered, the tritium arisings are the same, regardless of the actual level of B-10 in injected boron. The EPR does not use boron as a burnable poison, preferring gadolinium oxide instead. Nor is boron used in control rods. There is no tritium production, therefore, from either burnable poisons or control rods. The boron concentration in the primary coolant can be reduced by increasing the use of burnable poisons. The EPR considers this and uses gadolinium oxide, which is mixed in with the UO_2 in some of the fuel pellets. However, it is not possible to use gadolinium oxide in all fuel since the mixture has a lower thermal conductivity than ordinary UO_2. Incorporating it into all pellets, therefore, has an economic penalty through a reduction of reactor power. The EPR design is optimised in respect of the quantity of gadolinium which may be utilised without power reduction. Increasing the design output to 4500 MWth required some changes in fuel management including an increase in the number of new assemblies loaded to conserve the same natural cycle length. The tritium source term is minimised by optimising the gadolinium load (PCER Sub-chapter 6.3, section 6.2.1). There is no general quantification as the reduction depends on the fuel management approach adopted.

BAT FORM FOR TRITIUM (continued)

3.2 Lithium

Lithium is injected to control reactor pH. It is required to counter the effects of boric acid injection and to minimise, subject to other constraints, the corrosion of the materials in the coolant circuit. Excess corrosion has financial and radiological implications and the coolant chemistry regime will be aimed at minimising both of these aspects (see PCSR Sub-chapter 5.5).

At the expected lithium concentrations in the primary coolant (a few ppm), with natural lithium, tritium production would be approximately 1 TBq - 2 TBq per day. This would clearly lead to very large arisings of tritium. It was decided, therefore, at an early stage in PWR design to deplete the Li-6 in the injected lithium to reduce tritium discharges. This is an inherent design feature, but is also an important source minimisation technique. The specification for the EPR is that Li-6 in injected lithium should not exceed 0.1%. Use of lithium hydroxide containing more than 99.9 atom % Li-7 minimises the production of tritium from the Li-6.

The level of lithium required is a function of the concentration of boron in the primary coolant. The approach adopted for the EPR is to enrich the boric acid so that the equivalent B-10 level, needed for reactivity control, is around 37% of the boron, as opposed to the natural level of 20%. This has the effect of reducing the level of boric acid, and hence lithium, in coolant. Optimal pH control by boron-lithium coordination ensures the integrity of primary circuit components (corrosion etc) including Steam Generator tubes, but at the same time avoids higher concentrations of Li that could act as a significant source of tritium (or give rise to other operational problems).

To offset the drop in the pH at 300°C due to an increase in the boron concentration inherent in the EPR fuel cycle of at least 18 months, a lithium concentration of 6 ppm at the beginning of fuel life is envisaged by applying B-Li ratios. However, it should be noted that control of the chemistry of the primary coolant in a PWR is constantly under review. Because of this, it is possible that by the time the EPR is operational, the composition of the primary coolant may have changed from what is currently under consideration. However, the fundamental design objective of the EPR is to minimise tritium arisings through optimisation of boron and lithium concentrations (see PCSR Sub-chapter 5.5).

3.3 Deuterium

Deuterium is a natural radioisotope found in water. It is possible to reduce its concentration by isotopic separation but the quantities of water involved are such that this is not a practical proposition. The quantity of tritium produced from deuterium is relatively small and therefore no design changes have been made to minimise this source.

BAT FORM FOR TRITIUM (continued)

3.4 Secondary Neutron Sources

The secondary neutron sources add significantly to tritium releases. The quantity of tritium they produce is proportional to the mass of beryllium they contain. The sources in the EPR have approximately half of the mass of beryllium in the sources at Sizewell B so measures have already been taken to minimise the tritium arisings.

It may be possible to totally remove the sources in the longer term. However, a new core will probably still need primary sources (which are likely to be of the antimony-beryllium type) so tritium production cannot be totally eliminated.

It might be possible to use alternative neutron producing sources, at the time when sources are needed in the core. The practicality of this could be examined but it is not clear that such sources are currently available or could be manufactured to the standard required for an EPR. Furthermore, to adopt a new type of source is a move away from an established and tested design. The current use of secondary neutron sources is considered to be consistent with industry best practice.

Finally, if the beryllium sources are retained, their stainless steel cladding could be replaced by Zircaloy which, as has been seen for fuel, is an effective barrier for retaining tritium. However, these sources would require replacing every five years. As well as the expense involved in the cost of the sources, there could be additional operator exposure as a consequence, and increases amounts of radioactive and conventional wastes with their associated disposal costs. There would also be some risk involved because this is an unproven design. This option for minimising tritium releases has therefore not been progressed for these reasons.

3.5 Fuel and Helium in Fuel Rods

Operation of the reactor will produce tritium through the fission process and cannot, therefore, be controlled. However, as discussed in section 1.2, the Zircaloy clad is a very effective barrier in preventing the release of tritium such that releases from the fuel are very low compared to other sources.

The design of the fuel assemblies means that the tritium activity in a single fuel pin at the end of its irradiation is around 0.1 TBq. Individual fuel can leaks will not contribute significantly to tritium discharges from an EPR, as noted in section 1.

Production of tritium from helium is a small source. It is far less significant than the fission process and as such does not require further consideration.

BAT FORM FOR TRITIUM (continued)

4. Demonstration that the techniques used for treating each process stream minimise the overall impact of the disposal of radioactive waste (including evidence of the consideration of disposal via different pathways).	There are currently no practicable options for the abatement of tritium before discharge, whether liquid or gaseous. The totality of tritium produced is discharged at some point as either a liquid or gaseous effluent. The volumes of liquid tritium discharged could be reduced by recycling. However, this would raise the concentration of the tritium in the primary coolant. It is essential to limit the tritium concentration in the primary coolant for radiological protection reasons in order to minimise tritium gaseous discharges (and thus inhalation doses to workers) both when opening the reactor core and in the vicinity of the storage pools (as discussed in section 1). The doses associated with tritium discharges show that the dose per unit release from gaseous discharges is far higher than that from tritium in liquid effluent. The optimum, given that discharges will occur, is to maximise liquid discharges. **4.1 Treatment of Liquid Effluent** Because of the way it is produced, most tritium resulting from the operation of a PWR is present in the liquid form as a result of primary coolant let-down. There is no specific process for the removal of tritium from let-down for routing as a gaseous or solid waste arising. As such, the majority of tritium will be discharged in the liquid effluents (> 95%) which results in the lowest radiological impact compared to gaseous emissions (see section 4.3). Management of liquid effluents and operational factors can lead a site to discharge more tritium at a given time of the year than at others, for example at the start of the fuel cycle. Some sites have reduced the tritium activity of the primary circuit as far as possible and therefore the Coolant Storage and Treatment System TEP [CSTS] distillates have been transferred to storage tanks prior to subsequent discharge. This management method is used on a regular basis on existing sites in order to comply with the radiochemical and radiation protection objectives for the primary circuit, and in particular is used prior to reactor shutdown in order to reduce the tritium activity of the primary circuit (PCER Sub-chapter 6.3, section 6.2.2). French nuclear power plants (NPP) have specific operational guidelines used to find a balance between tritium production in the reactor coolant system and the amount which could be potentially discharged. They have at their disposal decision making tools and information to facilitate site decision making on implementing a tritium management strategy [DEAPC080379 Revision A1, “Assistance and Recommendations for Tritium Management in NPP Systems].

BAT FORM FOR TRITIUM (continued)

	In the event of small primary to secondary leaks (or more major tube failures), the blowdown water from the steam generators may contain low levels of tritium from the primary coolant circuit. This effluent source is normally directed to the Steam Generator Blowdown System (APG [SGBS]) where it is filtered and demineralised, then recycled to the main turbine condenser. Exceptionally, when recycling is not possible, the blowdown is sent to storage tanks before monitoring and discharge (PCER Sub-chapter 6.2, section 1.1.2). The site and utility specific detailed management of liquid discharges is not determined at the GDA stage. Section 6.2.2 of Sub-chapter 6.3 of the PCER states that the current French fleet management policy (and is also the strategy at Sizewell B) is continual let-down of all tritium produced in order to avoid an increase in the concentration of tritium in the primary circuit and the associated radiological protection issues. Discharges can be reduced temporarily by recycling the Coolant Storage and Treatment System (TEP [CSTS]) distillates. This means that tritium is concentrated in the circuits instead of being discharged. This management technique can only be used on a temporary basis as it is necessary to limit the tritium activity in the primary circuit. However, this method can be used if, for a particular reason, the discharges need to be temporarily reduced (for example, in the case of a non-coastal site, if the river flow rate does not allow tanks to be emptied). Conversely reduction of the tritium activity of the primary circuit will temporarily increase tritium liquid discharges. The strategy for management of tritium will also have an impact on the tritium concentration in the storage and discharge tanks. For instance, Chooz site OPEX from 1999 to 2011 show high variations in the tritium concentrations contained in the T (0KER [LRMDS]) tanks: the average tritium concentrations are approx. 1.79 MBq/l and the maximum values reported were approx. 7.9 MBq/l. These variations are mainly due to the way tritium has been managed from 1999 to 2011. Overall, and considering the enhanced power production of the EPR compared to Sizewell B, it is evident that the EPR presents an improved performance for the tritium liquid discharges normalised to power output. Details are given in section 3.3 of Sub-chapter 8.2 of the PCER. In particular, it was estimated by the Environment Agency [Generic Design Assessment of New Nuclear Power Plant Designs. Statement of findings following preliminary assessment of the submission by AREVA NP SAS and Electricité de France SA for their UK EPR design. The Environment Agency. March 2008] that the liquid discharges of tritium normalised to 1000 MW(e) are over 50% lower for the EPR than for Sizewell B (43,228 GBq y^{-1} estimated for the EPR, 67,230 GBq y^{-1} for Sizewell B). This reduction of tritium discharge for each MW(e) produced demonstrates the ability of the EPR to minimise the radioactive source term, and complies with the BAT principle of improving the eco-efficiency of the facility (i.e. reduced emissions per GW).

BAT FORM FOR TRITIUM (continued)

4.2 Treatment of Gaseous Effluent

Discharges of tritium in gaseous effluent are the result of evaporative losses from sources of tritiated water. Potential sources are:

- the in-containment refuelling storage water tank (IRWST) (except the part of the tritium which recondenses on the cold parts of the EVR [CCVS]);
- the spent fuel storage pool;
- at a lesser extent, tanks of primary coolant effluent treatment systems.

Primary gaseous effluent is processed in the Gaseous Waste Processing System (TEG [GWPS]). The recombination unit may help to ensure that tritium in the purge gas in the TEG [GWPS] is returned to and retained in the liquid phase, although to date this effect has not been quantified (PCER Sub-chapter 6.2, section 1.2.3). Evaporative losses from the IRWST and the spent fuel storage pool are discharged to atmosphere through the stack except the part which condenses on the cold parts of the EVR [CCVS].

Small leaks may occur between the primary and secondary circuits through which tritium leaks and appears in the secondary circuit and condensed secondary water. With low pressure in the condenser wet well during operation, some tritiated water can therefore appear in the main condenser off gas. This is collected in the Condenser Vacuum (CVI) system, and then sent to the Nuclear Auxiliary Building Ventilation System (DWN [NABVS]), where it is passed through a HEPA (High Efficiency Particulate Air) filter before being discharged into the stack (PCER Sub-chapter 6.2, section 1.2.2). Leaks will be minimised through the use of suitable materials and construction techniques and appropriate maintenance.

All radiation controlled areas of the Nuclear Auxiliary Building, the Fuel Building, the Safeguard Buildings, the Reactor Building, the Operational Service Centre, the Access Building and the Effluent Treatment Building are served by dedicated HVAC systems. These ensure pressure differentials and air changes are maintained so that air always moves from potentially less contaminated areas to more contaminated areas, in accordance with standard best practice (PCER Sub-chapter 6.2, section 1.2.2). These HVAC systems incorporate redundancy hence reduce evaporation of tritium.

Evaporation of tritium can also be reduced by the cooling systems that control the temperature of the IRWST, the reactor cavity during shutdown and the fuel pool liquid and gas blanket.

BAT FORM FOR TRITIUM (continued)

Section 1.2.1 of Sub-chapter 6.2 of the PCER states that there are no cost effective methods for abatement of tritiated water vapour and the BPEO at all nuclear power plants is to discharge most via the stacks serving the various areas where tritiated water vapour arises.

With respect to gaseous effluent discharges, changes have been implemented to minimise gaseous arisings that contain tritium. In the current 1300 MW(e) French PWR designs, there is intermediate flushing of the primary Coolant Storage and Treatment System (TEP [CSTS]) tank that is the source of approximately 80% of the tritiated gaseous effluents in discharges from these plants. To minimise this source of tritium, the EPR uses the alternative N4 system for the collection and treatment of primary circuit coolant as this is let-down from the circuit over the operating cycle. As a result, this source of tritium is removed, and in the EPR the bulk of the tritium in gases originates from evaporation from the fuel storage pools and the IRWST. Evaporation from the fuel storage pool is predicted to result in a loss of 0.35 TBq y^{-1} and that from the IRWST up to 0.5 TBq, with a realistic value somewhat below this.

The annual expected performance of the EPR for the discharges of gaseous tritium represents a decrease of 60% compared to the 1300 MW(e) units per MWh produced.

Operating experience feedback at Sizewell B reported in 2005 that the average twelve-month discharge for gaseous tritium was around 0.8 TBq, and that a maximum of 1.9 TBq y^{-1} had been observed, in comparison with a discharge limit of 8 TBq y^{-1}. Values normalised to 1000 MW(e) power output provide evidence of the improved performance of the EPR; the EPR gaseous tritium discharges are expected to be over 30% lower than at Sizewell B (2520 GBq y^{-1} expected at Sizewell B vs. 1730 GBq y^{-1} expected for the EPR, normalised at 1000 MW(e) [Generic Design Assessment of New Nuclear Power Plant Designs. Statement of findings following preliminary assessment of the submission by AREVA NP SAS and Electricité de France SA for their UK EPR design. Environment Agency, March 2008].

4.3 Discussion of the Minimisation of the Radiological Impacts of the Tritium Discharges

In order to consider the minimisation of impacts of tritium discharges it is necessary to consider the radiological impacts in terms of radiation doses. These are discussed in detail in Sub-chapter 6.3 and Chapter 11 of the PCER. The methodology for the detailed radiological assessment was based on the use of PC CREAM 98 software tool and is described in Sub-chapter 11.1, section 1.3 of the PCER.

BAT FORM FOR TRITIUM (continued)

Doses to maximally exposed individuals

The Tritium maximum annual liquid (75 TBq y^{-1}) and gaseous (3 TBq y^{-1}) discharges for an EPR unit were used for the assessment. The table below sets out the calculated doses to the maximally exposed individuals from both liquid and gaseous discharges.

Doses to maximally exposed individuals from Annual Releases of tritium in gaseous and liquid effluents

Maximally exposed individuals	Exposure from gaseous discharges (3 TBq y-1) (μSv y^{-1})	Exposure from liquid discharges (75 TBq y-1) (μSv y^{-1})	Local resident exposure from both liquid and gaseous discharges (μSv y^{-1})
Adult	0.14	0.02	0.16
Child	0.14	0.005	0.15
Infant	0.26	0.002	0.26

It is theoretically possible for a family (adults, children and infants) living in the nearest habitation to be exposed to both atmospheric discharges and to liquid discharges in the marine environment. To determine the dose to these local residents it is necessary to sum the doses from aerial discharges and liquid discharges, and these are shown in the above table for completeness. The highest total dose calculated to a local resident (adult) from exposure to all radionuclides discharged from an EPR is 21 μSv y^{-1}. As such, and even though the proposed maximum discharge values for gaseous and liquid tritium may be seen as high in numerical terms, the discharges would only have a minor radiological impact on the overall dose received by the most exposed members of the public.

BAT FORM FOR TRITIUM (continued)

Collective Dose

The collective dose is a measure of the radiation exposure in a population and is the sum of effective doses from a given practice or situation to all affected individuals, now and in the future. The environment agencies guidance on prospective doses assessment recognises 500 years as an appropriate measure for collective doses assessments.

Collective doses to the populations of UK, Europe and the World, truncated at 500 years, are estimated and presented in PCER Sub-chapter 11.1, section 4. As the site location is an important factor for estimating the dose to the UK population, it is necessary to make some assumptions about the power station location. Among the potential sites where a new EPR reactor could be located, the site which presents the maximum collective dose is presented in PCER Sub-chapter 11.1, section 4.2. The estimated collective doses are presented in the table below.

Collective Dose from Annual Releases of tritium in gaseous and liquid effluents

		UK	**Europe**	**World**
Collective Dose due to tritium (Man Sv)	**Atmospheric discharges**	2.3×10^{-3}	7.9×10^{-3}	9.8×10^{-4}
	Liquid discharges	3.2×10^{-5}	2.9×10^{-4}	3.3×10^{-3}
	Total	2.3×10^{-3}	8.2×10^{-3}	4.3×10^{-3}
Total Collective Dose (Man Sv)		0.31	2.46	16.9

The collective dose from atmospheric and liquid discharges is dominated by releases of C-14, accounting for almost 100% of the dose for all populations considered. The contributions to the collective dose from discharges of tritium to UK, European and World populations are very small.

BAT FORM FOR TRITIUM (continued)

Potential Short-Term Dose

Potential short-term doses have been assessed for gaseous discharges. For the radiological impact assessment of gaseous discharges, it is normally assumed that discharges occur continuously and uniformly over a year. However, during normal operations, short-term gaseous discharges can occur during, for example, outages and start-up or when purging the cooling system. It is possible that such short-term discharges may lead to doses that are higher (or indeed, lower) than would be expected if it were assumed that the discharges are continuous over a year.

The methodology used and calculated doses are presented in PCER Sub-chapter 11.1, section 2, using the release figures for C-14 presented in section 2.2 of Sub-chapter 11.1 of the PCER. The table below shows the dose to adults, children and infants from all pathways and radionuclides as a result of a single short term discharge. Despite the conservative assumption used, which is that, all radionuclides are discharged together in the same release; the total short-term dose is less than the total continuous release dose. The contributions to the potential short-term dose from gaseous discharges of tritium to these exposure groups are very small.

Potential Short Term Doses

Radionuclide Group	Effective Dose per discharge (µSv) for a 24 hour scenario		
	Adult	Child	Infant
Tritium short-term dose	0.020	0.022	0.044
Total short-term dose	0.864	0.889	1.46

BAT FORM FOR TRITIUM (continued)

5. Demonstration that the techniques used to minimise impacts are BAT by reference to all other options considered and rejected or by benchmarking against best world practice.	**5.1 Liquid Tritium** A number of techniques have been developed, primarily for tritium removal or the detritiation of heavy water (for Heavy Water Reactors). These techniques include electrolysis and catalytic exchange, and fractional distillation. Tritium removal from light water, however, requires larger stripping and recovery factors than for heavy water. To be capable of obtaining the required high recovery, a feasible tritium removal process would require a high isotopic separation factor. The IAEA in its Technical Report No 421 [Management of Waste Containing Tritium and Carbon-14 IAEA, July 2004] identifies the hydrogen-water chemical exchange process as having the required high isotopic separation factor and it is suited to a water feed. In addition, the technical feasibility of hydrogen isotopic enrichment through the exchange between hydrogen and water is established. IAEA Technical Report No 421 describes a study undertaken in 1998 by AECL, commissioned by the USDOE, which examined the detritiation of light water containing very low concentrations of tritium. Of the technologies examined, six were judged to be uncompetitive or impractical: water distillation, electrolysis, bithermal ammonia hydrogen, monothermal ammonia hydrogen, laser isotope separation and water permeation. The processes that showed potential for possible development included chemical exchange: combined electrolysis and catalyst exchange (CECE), Girdler sulphide and bithermal hydrogen-water exchange. However, no information is available to suggest that any of these have been progressed and used anywhere on an operational PWR. Section 2.2 of Sub-chapter 6.3 of the PCER concludes that there are currently no processes implemented on an industrial scale for the treatment of tritium in liquid phase for a PWR, and therefore all liquid tritium produced is discharged. In addition, because of the length of the half-life of tritium it is not possible to store all the liquid tritium produced for decay before discharge, as the volumes involved would be too large. This would also increase the risk of uncontrolled gaseous tritium discharge due to evaporation of the storage tanks. This conclusion agrees with that of the OECD [Effluent Release Options from Nuclear Installations Technical Background and Regulatory Aspects OECD, 2003].

BAT FORM FOR TRITIUM (continued)

It is possible to give more detail as to the validity of this conclusion:

- the half-life of tritium is over 12 years. Clearly, to obtain any significant reduction in tritium discharges requires retention for many years. The volume of the primary circuit of an EPR is over 200 m^3 and at least this volume will be discharged every cycle. In addition, Sizewell B discharges some 10,000 m^3 per annum of liquid waste, most of which is tritiated to some degree. This is via process drains, chemical drains and floor drains. Over a 12 month period, therefore, a considerable volume of tritiated water will have been discharged. This would require a large, double contained storage facility which would need to be monitored;
- it might, in theory, be possible to collect all discharged primary coolant in a smaller facility and re-cycle it. However, the tritium in circuit would simply accumulate. There would be some small benefit from decay but increased coolant levels could give rise to operator exposure issues;
- evaporation is not an option since the tritiated water would carry over with the ordinary water to be condensed so that, even though there may be some tritium left in the evaporator concentrates, most of it would be collected in the condensate tank in liquid form. This process also runs the risk of increasing the discharges of tritium in gaseous effluent and thus of the radiological impacts of tritium disposal.
- filtration is not relevant for abatement of tritium in liquid waste;
- the tritiated water could be cemented. This, however, would produce very large volumes of solid waste (probably ILW) which would need to be disposed of and it is not believed that this would necessarily immobilise tritium effectively;
- isotopic retention is a process by which selective migration of specific radioisotopes is catalysed by an electrochemical process. However, development of this mechanism is at an early stage and it is not clear how efficient it will be. Implementation will be expensive.

In summary, although theoretically there are treatment processes which could be used to reduce liquid tritium discharges from an EPR, the judgement is that the benefit in reduced public exposure from the use of such processes is grossly disproportionate to the cost and resources of their implementation (which would include the detriment of having to dispose of all of the associated secondary solid wastes). This position is strengthened by the fact that none of the treatment processes are currently used anywhere on an operational PWR and would need significant development work for such use. It is therefore concluded that the techniques used for disposal of tritium in liquid discharges from the EPR are the Best Available Techniques.

BAT FORM FOR TRITIUM (continued)

5.2 Gaseous Tritium

Similar conclusions apply to discharges of tritium in gaseous form. Potential processes to reduce gaseous effluent discharges are:

- collection and retention of effluent gases, to allow radioactive decay, is impracticable;
- filtration is not relevant for abatement of tritium in gaseous waste;
- the new build BAT assessment (see above reference) considers the possible use of molecular sieving in which gas is oxidised and the tritiated water which is produced collected by the sieve. The sieve is then disposed as solid waste or the water may be desorbed and discharged as liquid effluent;
- packed bed or plate columns can be used to counter-currently contact the gas stream with a chilled water stream. The degree of water removal from the system is not high, but the performance may be improved by an exchange of water vapour between the gas and liquid phases. This technique can only be used if the intention is to generate a dilute tritium stream for direct discharge. This would divert tritium from gaseous to liquid discharge, but given the very low ratio of tritium being discharged by the gaseous route (2% - 3%) and the inefficiency of the process, the benefits achieved would be grossly disproportionate to the costs of implementation;
- evaporation of tritium from the IRWST and the fuel pool depends on the ambient temperature and hygrometry. The higher the ambient temperature and hygrometry in the building are, the less gaseous tritium is evaporated from the IRWST and the fuel pool. Yet health and safety of workers in the building and operating conditions of equipments impose a maximal admissible temperature and hygrometry. Therefore, the temperature and hygrometry are optimised based on a compromise between working and operating conditions (which are a priority) and reduction of evaporation of tritium. The conditions are reached using the cooling systems for the fuel pool liquid and gas blanket and the use of HVAC systems.

On the basis of this discussion it is concluded that the techniques used for disposal of tritium in gaseous discharges from the EPR are the Best Available Techniques.

BAT FORM FOR TRITIUM (continued)

6. Demonstration that the techniques used conform to the IWS (cross-reference to IWS sections)	The management of wastes are described in the following sections of the Integrated Waste Strategy (IWS) document Section 5 Integrated Waste Management Strategy Section 5.2 Operational radioactive wastes Section 5.2.1 Solid wastes Section 5.2.2 Liquid radioactive wastes Section 5.2.3 Gaseous radioactive wastes. It should be noted that the IWS does not provide detailed discussion on tritium, which is discussed in this form.
7. Identification of disposals resulting from the use of techniques to minimise overall impacts and quantification of annual volumes and activity of nuclides.	**7.1 Solid Wastes** Given that tritium is highly mobile, it is likely to be present in all solid waste at a relatively low level. Tritiated solid waste would only become a potential issue if one of the specialist options for removing tritium from liquid or gas discussed in section 5 were to be implemented. **7.2 Liquid Discharges** The determination of monthly discharge values is difficult due to the nature of the tritium discharges that are very dependent on operating conditions and the future site management policy (which is beyond the GDA stage). Overall, the fluctuations over the whole fuel cycle are significant. If there is a continuous purge of coolant from the reactor, then tritium discharges will closely align with production and the limit would be controlled by the predictions of arisings in the first twelve months of operation in the cycle. On the other hand, coolant may not be continuously purged to the environment through the cycle but may be retained and discharged over the last few months of cycle operation. In this case, it is possible that up to two months' production of liquid tritium would be discharged in one month. Considering a worst case scenario where this maximum discharge of two months' production coincides with the maximum tritium production of the cycle, it was estimated that this would correspond to around 25% of the annual discharge released in a month. This leads to a maximum monthly discharge of 18.75 TBq per month. In the absence of contingencies, the monthly discharges are expected to be around 25% of the annual expected discharge, i.e. 13 TBq per month. The associated headroom is expected to cover for all contingencies, whether linked to operational events or uncertainties due to the lack of OEF, in particular concerning the fuel management mode used (PCER Sub-chapter 6.3, section 6.2.2).

BAT FORM FOR TRITIUM (continued)

Operational contingencies that may affect discharge levels include:

- unplanned shutdown. Since the production of tritium is almost proportional to the energy production, it is evident that large fluctuations during a fuel cycle are observed. In particular, the annual discharges of liquid tritium are much lower if a reactor has been shutdown during the year (either for refuelling or for unplanned shutdown) than if it has been operational throughout the year. In addition, variation over the fuel cycle also impact annual discharges; as tritium production is higher at the beginning of a cycle than at the end (higher boron and lithium concentration in the primary circuit), discharges are also generally higher in the first months of a cycle before tailing down. The starting point of a cycle in the year thus impacts the overall annual discharge (over a calendar year). This issue is not necessarily seen if the discharges are considered over a rolling year;
- fuel leaks, implying a higher tritium concentration in the primary circuit than usual and therefore higher tritium discharges;
- one (or several) 0KER [LRMDS] storage tank(s) become unavailable for use, for example following an unexpected contamination of the effluent, which cannot in turn be discharged.

In order to estimate the average discharges of tritium, the production coefficient (ratio of the number of EFPD operation to the number of effective days) considered is taken at 91%, the start of the cycle is randomly spread over the year and the median hypothesis of lithium concentration is at 3.5 parts per million (ppm) for UO_2-IO-18 month management. This leads to an estimate of 52 TBq y^{-1} of tritium discharged as a liquid**.**

To estimate maximum discharges, a production coefficient of 100% is used and it is given that the annual tritium production of a unit is maximised if its cycle starts on 1st January (maximum boron concentration). The Li concentration is conservatively considered at 6 ppm for UO_2-IO-22 month management. This leads to 75 TBq y^{-1} of tritium discharged as a liquid (PCER Sub-chapter 6.3, section 6.2.1).

7.3 Gaseous Discharges

The monthly profiles provide evidence that the gaseous discharges are not constant over a fuel cycle and that the highest discharges are over 15 times higher than the lowest monthly discharges. In addition, it is useful to note that most of these maximum values have been recorded when only one of the 2 units on site was operational, the other one being shutdown for maintenance. Thus, as a first approximation, it can be considered that these values are the maximum recorded over a month for one unit.

BAT FORM FOR TRITIUM (continued)

The proportion of tritium discharged in the gaseous form is difficult to assess and quantify as it can be largely influenced by operating conditions and the evaporation rate of the pools, and therefore monthly gaseous discharge profile will closely follow the profile of the power production. As such, it is expected that the higher monthly discharges will only represent a fraction of the total annual discharge. This was the case both at the French PWR units at Chooz and Civaux where the highest monthly discharges (both in June 2006) presented only a maximum of 1.5% and 2.0% of the allowed annual maximum discharge value of the site.

Potential short-term discharges are considered in the PCER for gaseous discharges only (PCER Sub-chapter 11.1, section 2.2). It is normally assumed that gaseous discharges occur continuously and uniformly over a year. However, during normal operations, short-term gaseous discharges can occur during, for example, outages and start-up or when purging the cooling system. It is possible that such short-term discharges may lead to doses that are higher (or indeed, lower) than would be expected if it were assumed that the discharges are continuous over a year. Short term doses are discussed in section 4 of this form and were found to be small.

The expected performance without contingency and the proposed annual maximum discharge value for the EPR gaseous tritium discharges are 0.5 TBq y^{-1} and 3 TBq y^{-1} respectively, and have been estimated using operating experience feedback data from the 900 MW(e) and N4 fleet (under the same considerations mentioned in section 7.2). Indeed, it has been established (see above) that, considering the EPR specificities, the main source of gaseous tritium discharges from evaporation of the storage pools (as for the 900 MW(e) and N4 reactors). The expected performance without contingency has therefore been calculated based on the ratio of the pools area between the existing units and the planned ones for the EPR, assuming a tritium concentration in the pool of the Fuel Building similar to that of the existing units. Considering that the source term is relatively well understood, the lowest annual maximum discharge value for the existing units (3 TBq y^{-1} for N4 and 900 MW(e) reactors) should also be applied for the EPR (PCER Sub-chapter 6.3, section 7.2.1).

3.4 BAT FORM FOR COBALT-58 AND COBALT-60

1. Mechanism of production of the radionuclide and how it relates to the process by which the nuclide is produced.	**1.1 Reactor Materials** The presence of corrosion products in the primary coolant is the result of complex processes, which involve interacting physical and chemical mechanisms on in-core reactor materials. Contact between primary circuit materials with water at high temperature causes the uniform and global corrosion of the metallic components of the primary circuit. It produces an oxide layer on in-core surfaces, especially on the surfaces of the steam generator tubes. Corrosion products are released or dissolved as ions in the coolant water, some of which attach themselves to particulate material or remain in solution. These particles are formed and can either be deposited on the surface of the circuit, or liberated by erosion of deposits. Those corrosion products are transported in the primary coolant and, when they set in or pass through the core area, they become activated by the neutron flux. Activated products can also be released in the primary circuit by corrosion and erosion of activated structural reactor components. Co-58 and Co-60 are corrosion products formed respectively by neutron activation of Ni-58 and Co-58, Co-59 being generated by deactivation of Co-58: • Ni-58 (n, p) -> Co-58; • Co-58 (n, γ) -> Co-59; • Co-59 (n, γ) -> Co-60. **Formation of Co-58 by activation of nickel-based alloys** Co-58 has a relatively short half-life of 71 days and is a beta and gamma emitter. It is produced by neutron activation of Ni-58, a major constituent of the steam generator tubes (made of 690 alloy) and of the stainless steel in core and vessel materials. As described above, steel activation products are released into the coolant by a number of mechanisms including corrosion.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

Formation of Co-60 by activation of stellite

Co-60 has a half-life of 5.27 years and is a beta and strong gamma emitter. It is produced by neutron activation of Co-59, a major constituent of reactor steel components, particularly stellite. Stellites are cobalt–based alloys with hard-facing characteristics. However, when a stellite-coated surface becomes worn out, particles are likely to detach from it and flow within the primary liquid towards the reactor core, where the cobalt content of the alloy becomes activated, producing Co-60 (strong gamma emitter) to further deposit as “hot spots” or transfer from the primary circuit as a liquid waste.

Stellites are currently principally used in PWRs in taps and valve seats, and in some parts of the internal core support structure and primary pumps. The design stage of the EPR has taken particular care to reduce the use of stellite (see section 3 below).

1.2 Oxygenation Peak

During refuelling outages, the primary coolant is oxygenated by the addition of hydrogen peroxide. The chemical conditions in the primary circuit therefore change from reducing to oxidising, resulting in a phenomenon known as a “crud burst” oxygenation peak (PCER, Chapter 8.2, section 3.3.1), [SZB/THR/042 - Review of the Control and Impact of the Discharge and Disposal of Radioactive Waste at Sizewell B Power Station (Information provided by British Energy Generation Limited for the review by the Environment Agency of authorisations under RSA 93)]. The switch to oxidising conditions causes much of the oxide layer formed during operation on the out-of-core primary circuit surfaces to be stripped off and either suspended or dissolved within the reactor coolant. This causes a temporary increase in primary coolant activity and consequently, radiation dose rates around RCP [RCS] components, but this is reduced rapidly by removing the “crud” in the filters and demineralisers of the RCV [CVCS]. Nevertheless, there is a slightly elevated activation product concentration in the reactor coolant following the crud burst, which leads to more radioactivity entering the 8TEU [LWPS]. If forced oxygenation with hydrogen peroxide were not carried out the crud burst would still occur, but in an uncontrolled way, when the reactor pressure vessel head is removed exposing the coolant to air.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

	1.3 Impacts on discharges and waste disposal Through reactor operational phases, Co-58 and Co-60 build up in the primary coolant and are therefore a problem if there are leaks and when components are opened up for maintenance. They spread to other systems (e.g. 8TEU [LWPS]) when maintenance is performed and are deposited on all the internal surfaces of these systems. They are both major constituents of radioactive liquid wastes and even after processing they can be measured in liquid effluent. Co-58 and Co-60 are key radionuclides to consider with respect to acceptability of solidified waste (e.g. ion exchange resins and filters) as they contribute together up to 80% to the dose associated with a waste package (PCER Sub-chapter 6.3, section 3.2.3).
2. i. Source term (before application of techniques to prevent or minimise at source have been applied), and; ii. Activity in reactor following the application of techniques to prevent or minimise the source term.	i. It is judged that quantitative data on the theoretical primary source term data is not required. In practice, producing robust data would be difficult due to the dynamic and complex processes involved with corrosion product generation and variability in residence time of the material within the reactor. It is considered more appropriate to demonstrate the measures taken to minimise corrosion products (which are discussed in section 3 of this form). Waste minimisation at source is achieved by core design, use of materials, primary coolant chemical and radiochemical specification and reactor operation. Plant performance for waste minimisation is, ultimately, measured by the volumes and radioactive content of all solid, liquid and gaseous disposals and discharges for an individual reactor. ii. In stabilised operation, the specific concentrations in the primary system of Co-58 is around 21 MBq t^{-1} as a realistic value and 390 MBq t^{-1} is the maximum value (PCER Sub-chapter 6.1, Table 1), based respectively on the average and maximum values recorded by gamma spectrometry for the N4 fleet in France. For Co-60, 2.3 MBq t^{-1} is considered to be a realistic value and 170 MBq t^{-1} is the maximum value in the primary coolant (PCER Sub-chapter 6.1, Table 1), under the same considerations. At the oxygenation peak that occurs at reactor shutdown, the specific concentrations in the primary system of Co-58 increases to 160 GBq t^{-1} as a realistic value and 250 GBq t^{-1} as the maximum value (PCER Sub-chapter 6.1, Table 2), based respectively on the average and maximum values recorded by gamma spectrometry for the N4 fleet. For Co-60, 3.3 GBq t^{-1} is the realistic value and 5.9 GBq t^{-1} the maximum value in the primary coolant (PCER Sub-chapter 6.1, Table 2), under the same considerations.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

	Note that quantification of the RCP [RCS] source term, based on calculations and considering specific materials and chemistry conditions of UK EPR [Analysis of UK EPR™ Source Term: Identification, Quantification and Characterisation. ECEF110448 Revision A. EDF. July 2011. (E) and Corrosion product characterization under PWR/EPR primary coolant conditions: thermodynamic assessments and power plant feedback. ECEF111022 Revision A.EDF. July 2011. (E)] shows the consistency of the nuclide source term specified in the PCER Sub-chapter 6.1.
3. Demonstration that the techniques used to minimise at source are BAT by reference to all other options considered and rejected or by benchmarking against best world practice.	**3.1 Minimisation at Source – Introduction** The hierarchy of measures featured in the EPR design to minimise at source the production of Co-60 and Co-58 activated corrosion substances in reactor primary coolant are described in the GDA submission, whose contents are summarised in the following paragraphs including, where necessary, supporting references. The main aspects to be considered for reducing the EPR radioactive source term are the selection of in-core reactor materials and their manufacturing processes, and primary coolant chemistry at the various stages of reactor operation. The evolution of the EPR design has benefited from decades of LWR research, development and operating experience feedback in France and, internationally, through participation and collaboration with other LWR operators and academic organisations. The principal drivers for optimisation of the reactor design with respect to source term reduction include worker ALARA/ALARP objectives and the need to reduce environmental impacts and solid radioactive waste liabilities. There are significant operational benefits to be gained from minimising the production of solid, liquid and gaseous radioactive wastes in addition to ensuring that the radiological impacts from waste disposals and discharges are minimised. For Co-60 and Co-58, the fundamental means of reducing their production in the reactor is to select carefully the specification for reactor component steels and reduce the use of Stellite, a cobalt-chromium alloy. The techniques used to minimise at source are discussed in the following paragraphs. **3.1.1 Stellite** As covered in section 1, stellite can be used for taps and valve seats in systems handling primary coolant, internal reactor core support structures and reactor coolant pumps. Stellite, in particulate and ionic form, becomes detached due to material erosion and corrosion processes, transferring to the bulk primary coolant resulting in elevated Co-60 concentrations. ‘Crud’ formation in the reactor circuit captures some of the Co-60 on internal surfaces and some of the activity remains in the primary coolant until removed by filtration and ion-exchange in the RCV [CVCS].

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

Stellite materials for taps and valves in primary coolant systems will be excluded from the EPR design and replaced with an alternative cobalt-free material. This design feature has been estimated to reduce the radioactive source term for Co-60 by 8% (PCER, Sub-Chapter 8.2, section 3.3.1). Development work to reduce the use of Stellite for other reactor components is on-going and significant reductions are anticipated based upon operating experience feedback at other PWRs where Stellite use has been reduced in major item replacement components such as reactor coolant pumps.

3.1.2 Limiting the Cobalt Content of Materials

Efforts are being made to ensure that the cobalt content of all materials will be minimised according to ALARP principles. A materials specification has been developed which is used for existing sites and the EPR [Radiation protection guidance for the design of pressurised nuclear equipment for French PWR reactors, ENRE06109C, E235/006341/CODNPRMN, Faidy.C, 07/09/2007]. The specification limits the cobalt content for steam generator tubing, bulk austenitic stainless steel, welding materials and other miscellaneous steels. It applies to those materials which come into contact with primary coolant at operating pressure and temperatures, or are used in primary coolant injection systems.

3.1.3 Steam Generator Tubing – Material and Manufacture

Improvements to the corrosion resistance of steam generator tubing materials are being developed and the 690 alloy is the chosen material due to the overriding requirement for minimising the possibility of Primary Water Stress Corrosion Cracking (PWSCC) (PCER Chapter 8.2 section 3.3.1). 690 Alloy is the chosen material for replacement of steam generator tubing on existing LWRs and can readily be supplied with an average cobalt content of less than 150 ppm [EPRI, Radiation Field Control Manual, 1003390, Final Report, December 2004].

As part of the manufacturing process, assessments are under way regarding the polishing of the steam generator channel heads in order to minimise the potential for corrosion product bonding, surface erosion and corrosion of the parent material, which, for the latter two aspects, would transfer into the primary circuit and thereby increase activation product generation. Electro-polishing of steam generator channel heads will be an operator specific choice and the assessment of the optimal arrangements for undertaking this during manufacture (type and length of treatment) will be chosen taking into account the cost and schedule.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

3.1.4 Pre-Operation Oxidation of Primary Circuit

During the plant commissioning programme, an oxide layer will be produced on reactor circuit components by exposing the material to demineralised water at high temperatures for a prolonged period in alkaline and reducing conditions. The entire primary coolant circuit will be flooded as part of the hot functional test. A detailed specification for this process is being developed to ensure that use is made following an ALARP approach based on the balance between the expected benefits and the tolerable risks in terms of safety, radioprotection and wastes as well as the operating experience feedback (particularly at EDF-N4 units, Sizewell B and Tomari 3).Prior to the hot functional test, flushing of the primary circuit, by purification using appropriate chemistry conditions and operation of the primary coolant treatment plant (RCV [CVCS]), will be undertaken (see PCSR Sub-chapter 5.5).

3.2 Primary Coolant Chemistry

See PCSR Sub-chapter 5.5.

The achievement of optimised primary circuit chemistry is one of the fundamental operational objectives associated with minimising the production of activated corrosion products at source. The selected chemistry regime for EPR balances the conflicting requirements for maximising fuel performance, fuel-clad integrity and primary circuit pressure boundary integrity, whilst minimising worker doses and radioactive waste generation. All PWR operators operate with primary coolant chemistry regimes that are cognisant of these requirements, which have been developed through the collaborative efforts of many national and international organisations as EPRI, INPO, NEI and the support of reactor designers.

Technical understanding is mature but industry experience continues to identify new challenges and benefits related to primary coolant chemistry management. It is prudent for AREVA, as the reactor designer, to maintain an on-going, open dialogue with industry peers and organisations to ensure developments not addressed in existing industry documents are accurately assessed for applicability to individual stations, including the UK EPR.

Industry-wide collaboration has resulted in the publication by EPRI of PWR Primary Water Chemistry Guidelines, “Primary Guidelines”, whose recommendations have been informed significantly by AREVA and EDF knowledge and experience [EPRI, Primary Water Chemistry Guidelines, 1002884, 2005]. EDF has developed its own primary chemistry guidelines for operating PWRs in France.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

	The EPRI guidelines are kept under routine review by an industry-wide committee facilitated by EPRI and supported by NEI and INPO [EPRI, 2005 Interim Review of the Pressurised Water Reactor Chemistry Guidelines – Revision 5, 1009933, Final Report, December 2005]. The current programme of work is diverse and the overall initiative is referred to as RP2020. RP2020 initiatives are currently being evaluated for EPR, which may be implemented for the UK EPR design. EDF has integrated the requirements for primary circuit chemistry into the EPR design process 'ALARP[1]' (Chemical and Radiochemical EPR Design Optimisation), working closely with AREVA. The output from the ALARP process is being integrated into the EPR design process. The chemistry regime currently selected for the EPR is consistent with those chosen for the current fleet of French PWRs, Sizewell B and many other PWR operators worldwide. Primary coolant chemistry may, however, be modified in response to developments arising prior to the start of reactor operations in the UK. Operating experience feedback and underpinning research has established that the proposed EPR chemistry regime is likely to be optimal for minimising corrosion products, with modifications to be implemented as necessary. There are three operational phases to be considered for ensuring that corrosion products are minimised at source. i. At Power A target of constant pH for primary coolant is proposed for the full duration of the power cycle. The PCER quotes a target value of pH 7.2, at 300°C, as this minimises the rate of crud deposition on in-circuit surfaces and, in particular, fuel element cladding (PCER Sub-Chapter 8.2, section 3.3.1.2.1.1). A higher pH is not advised as the lithium concentrations required would increase likelihood of corrosion of fuel cladding and other in-circuit materials, although this decision could be reviewed. The maximum in-circuit lithium concentration, at beginning of cycles, is 4 mg kg^{-1} for the EPR. It is anticipated that pH control will be automatic for the EPR using a recently developed lithium hydroxide auto-injection system. Trials at the Tricastin Unit 2 PWR operated by EDF have been successful and it is included in the EPR design. Moreover, zinc injection will be implemented in the UK EPR taking into account the expected benefits as presented below and detailed in PCER Sub-chapter 8.2.

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[1] Initially identified as CREDO approach: Chemical and Radiochemical EPR Design Optimisation.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

	UK-EPR conditioning Zinc Injection	BAT/ALARP consideration
	• Target Values : 5<[Zn]<15 µg/kg	• Reduction of material corrosion and replacement of cobalt on ex-core surfaces. No higher concentrations of zinc are requested due to the choice of material without Stress Corrosion Cracking risk and the use of zinc from the beginning of the operation. ➔ Reduction of corrosion products source term and reduction of fuel crud → reduction of nickel, iron, cobalt activation, hence reduction of Co-58, Co-60, Fe-55, Fe-59 production. This enables: o Reduction of cobalt and iron discharges o Reduction of solid wastes (filters and resins)
	• Use of depleted zinc acetate	• 1) The selection of depleted zinc instead of natural zinc is justified by: o Zn-65 limitation in the oxide layers of primary circuit surfaces in order to optimise the dosimetry benefit; o Zn-65 limitation in effluents and waste. 2) The selection of zinc acetate is justified by: o The high solubility of acetate when compared with other possible additives (borates, formates) which limits precipitation into the injection lines and allows easier staff chemistry activities concerning the maintenance of the injection system device. o International experience showing the absence of negative impacts on: - carbon production (negligible production 10^{-4}% of the carbon production from other sources [Zinc Injection Implementation at UK-EPR.- ECEF110138 Revision A. EDF. March 2011. (E); Zinc Injection claims, arguments and evidences: overall balance for UK-EPR - ECEF110139 Revision A. EDF. March 2011. (E)]; - waste production; - filter and resins consumption; and - CRUD deposition on fuel cladding: possible decrease of crud accumulation in the spent fuel.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

UK-EPR conditioning Zinc Injection (Cdt)	BAT/ALARP consideration (cdt)
• Maximum value: 40 µg/kg	• Conservative value to prevent crud

ii. During cold planned shutdown

A modified chemistry regime (reduced pH) is implemented at PWRs prior to refuelling upon shutdown, the primary purpose of which is to remove corrosion products from in-circuit and fuel cladding surfaces. This measure will primarily reduce worker doses. This standard practice reduces the radioactive source term by avoidance of further corrosion product activation and minimising the potential for fuel cladding corrosion and consequent leakage into the primary circuit.

The main stages to achieve these objectives involve reducing pH by removal of lithium in addition to forced oxygenation through introduction of hydrogen peroxide followed by high flow rate decontamination in the RCV [CVCS]. There is an expected temporary increase in filter arisings and secondary radioactive liquid effluent arisings as a result of these activities.

iii. During start-up

After refuelling, primary coolant must be degassed and purified to reduce impurities such as dissolved oxygen and nickel to low concentrations, which are specified according to requirements.

3.3 Best Practice and Summary

The design features of the EPR that ensure activated corrosion products are minimised are wholly consistent with other PWR designs and, to some extent, dependant on reactor age and design constraints of individual reactors.

Radiological protection standards for workers, solid waste management requirements and the ability to undertake maintenance and operate the reactor in a safe and efficient manner require the radioactive source term for Co-60 and Co-58 to be minimised as part of overall corrosion product minimisation.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

	In particular, whilst the materials used for the reactor and its ancillary components that contribute to Co-60 and Co-58 are specified to minimise the source term, the fundamental requirement is to select materials that ensure nuclear safety, are compatible with the range of conditions experienced in the reactor, and are economically acceptable.
4. Demonstration that the techniques used for treating each process stream minimise the overall impact of the disposal of radioactive waste (including evidence of the consideration of disposal via different pathways).	**4.1 Management and Abatement** Once generated, the primary technique for minimising Co-60 and Co-58 disposals and discharges are through operation of the reactor purification plant (RCV [CVCS]), which includes filtration and ion-exchange systems. The operation of the boron recycling system is also effective for removing corrosion products from primary coolant let-down, which are polished on a specific ion-exchange bed. These techniques minimise the specific activity of the primary coolant, a very minor proportion of which transferred to the liquid effluent treatment systems through managed let-down and minor leakage. Purification of primary coolant does not, however, achieve complete decontamination and removal of corrosion products, although high decontamination factors can be achieved. Co-60 and Co-58, amongst other corrosion products, are normally present in low concentrations in primary coolant let-down to the liquid radioactive waste effluent treatment system, where further purification is undertaken by a combination of evaporation, ion-exchange or filtration. Activities associated with the operation and maintenance of the primary and secondary liquid effluent systems and management of secondary wastes from effluent treatment will give rise to particulate (containing corrosion products) which are collected by the gaseous radioactive waste treatment system (HVAC system). Gaseous discharges are processed by filtration and, where required, iodine adsorption filters, to ensure discharges are ALARP. Direct discharges of beta particulate are very low and arisings of secondary wastes (in the form of HEPA filters and iodine adsorption filters) are considered to be acceptable based upon operating experience feedback. (See below for more detailed information on HEPA filtration). The objective is, therefore, to ensure that Co-60 and Co-58, amongst other corrosion products, are removed from liquid effluent at source and concentrated into solid waste by-products such as ILW and LLW spent filters, spent ion-exchange resins and evaporator concentrates from primary coolant and non-recyclable effluent treatment.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

Co-60 is a targeted radionuclide for abatement in liquid and gaseous effluents as it is a relatively significant contributor to public dose compared to other particulate activated corrosion products. Due to its half-life of 5.27 years, Co-60 in solid waste is retained long enough to allow sufficient decay before it reaches the environment from the landfill. If the solid waste is incinerated, Co would be either retained in the ash or transferred to the abatement plant secondary solid waste. Therefore, the overall radiological impact of Co-60 in solid waste is lower than the impact of the environmental discharge of liquid and gaseous effluents. Co-58, which has a much lower radiological significance, is minimised as a consequence of Co-60 abatement. Targeted abatement represents BAT and is consistent with the ‘concentrate and contain’ principles, provided that the solid waste volumes produced are not excessive. This is achieved through optimisation of the abatement process and implementation of best practice.

These are matters for the plant operator. The GDA submission states that the EPR allows maximum flexibility with regard to meeting expectations for the demonstration of BAT. Depending on decisions made by the prospective licensee in the UK, the EPR is capable of achieving very low radioactive liquid discharges for Co-60 and Co-58 as demonstrated by current French PWRs.

However, the achievement of low discharges will involve a number of disadvantages, in particular increased solid waste generation and worker doses mainly through the operation of evaporators and increased use of demineralisers. The solid wastes and other impacts arising from abatement are not considered to be grossly disproportionate to the benefits gained.

Particulate Removal in Liquid Effluents

The use of single-use cartridge filter technology has for some time been the preferred technique for particulate removal in primary coolant and radioactive liquid effluent systems for all PWRs [EPRI Radwaste Desk Reference, Volume 3, Part 1, Processing Liquid Waste, 2901 - 2902, May 1994] [EPRI, Radioactive Liquid Processing Guidelines, Technical Report, 1011728, November 2005]. Different types, brands and specification of cartridge filters can be used, depending on the characteristics of the liquid to be treated. There is considerable experience in the use of cartridge filtration for PWR liquid effluent treatment and the management of spent filters. From an operational perspective these are easier to apply than alternative filtration techniques (this technology is also more reliable relative to more complex filtration systems, due to its passive simplicity), many of which would require adaption to cope with conditions experienced in the primary circuit.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

	As stated in the PCER, the filters specified for use in the EPR primary coolant and liquid effluent treatment systems for the removal of corrosion products associated with particulate material have been principally selected on the basis of operating experience feedback at existing sites in France and Germany and in particular, for EPR, those particulate filters and ancillary systems used for the 1300 MW(e) units. It should be noted that there is no prescriptive filtration performance specification for these reactors and small modifications have been made to filtration strategies depending on individual reactor requirements. The filtration selection strategy for EPR is the result of many years of accumulated experience and knowledge and can be considered as evolutionary. This information is not readily summarised as it is based on historical information and decision-making over many years of reactor operation. However, this evolutionary approach has resulted in optimised worker doses, acceptable commercial impacts, and acceptable solid waste arisings whilst achieving low radioactive discharges from all reactors. The filters selected for the EPR vary in specification depending on the liquid treatment system considered. These choices are underpinned by guidance for the use of cartridge filters the fundamental aims of which are to provide best practice and advice to reactor operators in the fleet on the choice of filter materials, specification for filter performance and change criteria and to summarise the practical experiences of French sites. The EPR design focuses on minimisation at source and this is reflected in the choice of filtration systems for the RCV [CVCS] system to achieve primary coolant chemistry specification, which is set at 1 µm. Experience suggests that the use of filters below 1 µm in the RCV [CVCS] system can be problematic with respect to the generation of spent ILW filters whilst having minimal additional impact on reducing radioactive liquid discharges or worker doses. The advantage of using filters with filtration level below 1 µm has not been demonstrated on a large scale programme, to date. Spent RCV filters arisings are anticipated to be low – refer to section 7 below. Filtration in the RCV [CVCS] also seeks to minimise the generation of spent resin by use of pre-filtration to limit the degradation of the resin bed material and downstream filtration to avoid the introduction of fines as impurities in the primary coolant. Filtration is also used in the reactor coolant pumps to prolong the life of the seals and minimise the introduction of corrosion products from operation of the pumps. Changes to the RCV [CVCS] design would require significant resources, which do not appear to be necessary in light of the good performance of similarly specified systems on comparable PWR plant.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

	For other liquid effluents systems (e.g. TEP [CSTS], 8TEU [LWPS], SEK [CILWDS]) a range of filter pore sizes are used (up to a maximum of 25 µm) and pre-filters/strainers are employed to protect micro-filters and minimise the generation of spent filters. The filters used in these liquid effluent systems are also amenable to either volume reduction or incineration, which reduces overall solid waste disposals. Final filters for discharge are rated at 5 µm, which is the French legal requirement for final treatment, and this is also consistent with EA guidance for the treatment of liquid effluents from nuclear reactors [A6, Abatement of Radioactive Releases to Water from Nuclear Facilities, Environment Agency, 1999]. The filters are changed when the pressure difference reaches a set limit or when a set radioactivity level limit has been reached. This means that the filters are not changed at a regular frequency, but rather as and when required, hence the filters will be used to their full capacity and not wasted. This minimises solid waste arising associated with filtration and is part of the application of BAT. Ultimately, as experienced by individual reactor operators and anticipated to some extent for every EPR, modifications for filtration systems are expected to be required throughout the lifecycle of the plant as conditions change or through different operational practices. The operator will then have the option to change the filter pore size based on operating experience feedback. In the UK EPR filtration systems the filter inserts can be easily replaced, the pore size of the dead end filters selected and utilised in the UK EPR can be modified by the operator depending on specific requirements and operating experience feedback. The absolute filter pore diameter provides criterion for initial filter selection. The filter performance will, in addition to its specific design, also be influenced by throughput, type of particulate, particle size distribution, particle shape and more. The effective pore size of a dead end filter and its typical removal efficiency will change during its operation as it builds up a filter cake. While the original filter efficiency will remain constant the increasing thickness of the filter cake will itself act as filter media. Hence even Totally Suspended Solids (TSS) smaller than the nominal pore size initially selected will be retained within this additional filtering layer. In addition, the operating experience feedback shows that decreasing the initial pore size of the dead end filters will have no significant impact on TSS. The reason for this is the previously described filtration effect of the filter cake.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

A negative effect is, that selection of smaller filter pores will lead to rapidly increasing pressure drop across the filter (fouling) hence requiring a more frequent change of the filter media and consequently produce more secondary waste for disposal.

The main issues that arise are problems with excessive filter blinding due to excess particulate burden, the need for enhanced treatment to reduce in-circuit activity (to minimise source term and reduce worker doses) or to meet lower discharge targets. These changes have to be balanced against potential increases to spent filters requiring management as solid waste.

Beyond GDA

Final decisions to define the operation of the EPR, as presented in the GDA submission, with respect to primary coolant purification and non-recyclable radioactive liquid effluent treatment performance will need to be determined by prospective EPR operators. It is not considered appropriate to present in the GDA submission, the operational practices and specifications implemented by EDF in France without a robust evaluation of the constraints and requirements influencing these aspects in the UK. Such an evaluation is to be completed as part of the work to support discharge limits for an environmental permit application. Specification of the requirements for primary coolant boron recycling by evaporation, evaporation of radioactive liquid effluent and ion-exchange are likely to be the main decisions to be made by prospective operators.

Particulate Removal in Gaseous Effluent

The standard technique for the removal of particulate from gaseous effluents collected by high volume air-extract systems in nuclear installations involves the use of High Efficiency Particulate Air (HEPA) filters (efficiency 99.97% in accordance with the Technical Guidance [Technical Guidance Note A5, Abatement of Atmospheric Radioactive Releases from Nuclear Facilities, Environment Agency, 1998] [NVF/DG001 An Aid to the Design of Ventilation of Radioactive Areas]. Whilst HEPA filters are effective in reducing gaseous radioactive discharges of particulate, their principal function is to provide safety and environmental protection in response to an event involving the release of significant particulate material, such as a fire involving radioactive waste materials or a fuel handling event, where extraction of air for environmental discharge is required. The use of HEPA filtration for high volume air-extract systems in nuclear installations is considered best practice and will form part of the nuclear safety case for the UK EPR.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

	The potential for radioactive particulate is minimised at source and based on operating experience feedback. The radioactivity content on spent filter arisings is not generally expected to be significant, thereby indicating that measures taken to minimise particulate discharges at source are effective. HEPA filters will be changed, where practicable, upon meeting high differential pressure advised by the manufacturer, which will be detected through a site maintenance and monitoring schedule. Choices for the selection of HEPA filter manufacturer will be based upon a procurement specification consistent with relevant performance standards. HEPA filters are installed on all HVAC systems servicing the EPR nuclear island and Auxiliary Buildings (radiation and contaminated controlled areas) where there is a potential for radioactive particulate in extract air-streams. The Reactor Building incorporates internal particulate and iodine filtration systems (EVF) which can operate at power to reduce radioactivity in air without the need for continuous extraction, and help to maintain air pressures in relevant plant areas to control air flow and environmental conditions. Pre-filters designed to capture particulate that might damage HEPA filters are used and this is standard practice. Inlet HVAC filters are installed to minimise the presence of particulate in air subsequently extracted for discharge, thereby reducing the non-radioactive particulate burden on HEPA filters and maximising their operational life. The HEPA filter type chosen for each system will depend on the environmental and flow rate conditions anticipated but the vast majority of filters will be of standard design for general HVAC operation. The incineration of air filters should be possible depending on the environmental permit limitations and requirements for the incineration route selected. Where incineration is not possible, air filters will be amenable to volume reduction by shredding/compaction. These practices have a significant impact on the volume of air filters requiring disposal, which is often a significant proportion of the overall volume of solid LLW produced on a PWR. A circular HEPA filter design is proposed for EPR, which is consistent with recent installations at a number of nuclear facilities in the UK. The HEPA filtration systems for EPR will meet industry codes of practice for the testing, design and operation with particular reference to UK standards such as NVF/DG001 [NVF/DG001 An Aid to the Design of Ventilation of Radioactive Areas], which is considered to be consistent with comparable standards in France. It can be considered that the systems used for EPR are consistent with those used for existing installations in the UK and employed on most other nuclear installations.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

Radiological impacts of the disposals of Co-60 and Co-58.

In order to consider the minimisation of impacts of disposal and provide context it is necessary to consider the radiological impacts of the disposal of Co-60 and Co-58 in terms of radiation doses from the EPR. These are discussed in detail in Sub-chapter 6.3 and Chapter 11 of the PCER. The methodology for the detailed radiological assessment was based on the use of PC CREAM 98 software tool and is described in Sub-chapter 11.1, section 1.3 of the PCER.

Co-60 and Co-58 are not listed as separate radionuclides in the Sub-chapter 11.1 of the PCER. They are described as other fission or activation products (FP/AP) other than H-3, C-14, halogens and noble gases. The maximum annual gaseous (189 MBq) and liquid discharges (2.07 GBq for Co-58 and 3 GBq for Co-60, so 5.07 GBq in total) of Co-60 and Co-58 for an EPR unit were used for the assessment. The table below sets out the calculated doses to the maximally exposed individuals from both liquid and gaseous discharges (PCER Sub-Chapter 11.1 Annex 2).

Doses to maximally exposed individuals from Annual Releases of Co-58 and Co-60 in gaseous and liquid effluents

Maximally exposed individuals	Exposure from gaseous discharges (189 MBq) ($\mu Sv\ y^{-1}$)	Exposure from liquid discharges (5.07 GBq) ($\mu Sv\ y^{-1}$)	Local resident exposure from both liquid and gaseous discharges due to Co-58 and Co-60 ($\mu Sv\ y^{-1}$)
Adult	< 1 (0.0236)	3.165	3.189
Child	< 1 (0.0134)	0.513	0.526
Infant	< 1 (0.0126)	0.0557	0.0683

The overall radiation doses from the discharges of an EPR power station are well within the dose limit for members of public (1000 $\mu Sv\ y^{-1}$) and the UK government's dose constraint for a single source (300 $\mu Sv\ y^{-1}$).

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

	Collective doses are negligible for discharges of Co-60 and Co-58 due to their relatively short-half lives (Sub-chapter 11.1 Annex 6 Tables A and B). Short-term doses to the critical group have also been assessed (conservative models have been used) for Co-60 and Co-58 and are < 1 µSv for adult, child and infant (PCER Sub-chapter 11.1 Annex 4 Tables C, D and E).
5. Demonstration that the techniques used to minimise impacts are BAT by reference to all other options considered and rejected or by benchmarking against best world practice	As stated in section 4, the techniques presented in the GDA submission and their capabilities, as outlined above, are typical for PWRs operating worldwide and can be considered as BAT. The application of these techniques is flexible depending on operator requirements and very low effluent discharges can be achieved, as demonstrated in France and other countries. Alternative techniques to filtration, ion-exchange and evaporation (currently proposed for the EPR) have been evaluated. Processing options are outlined in the Solid Radioactive Waste Strategy Report [NESH-G/2008/en/0123/Rev. A - Solid Radioactive Waste Strategy Report] and PCER (PCER, Sub-Chapter 8.3, section 3.3.3). None of these options are currently considered in the GDA reference case based on Flamanville 3. However, these options will be kept under evaluation for potential future application. These options include ultrafiltration, cross-flow filtration, electrolysis and isotopic retention using electrochemical methods. Further evaluation of alternative techniques will, additionally, become a responsibility for prospective operators to meet environmental permit commitments for the demonstration of BAT.
6. Demonstration that the techniques used conform to the IWS (cross-reference to IWS sections),	The management of wastes are described in the following sections of the Integrated Waste Strategy (IWS) document: Section 5 Integrated Waste Management Strategy Section 5.2 Operational radioactive wastes Section 5.2.1 Solid wastes Section 5.2.2 Liquid radioactive wastes Section 5.2.3 Gaseous radioactive wastes. It may be noted that the IWS does not provide detailed discussion on Co-60 and Co-58, which are discussed in this BAT form.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

7. Identification of disposals resulting from the use of techniques to minimise overall impacts and quantification of annual volumes and activity of nuclides	The information in this section comprises information extracted from the PCER, Sub-chapter 6.3. Refer to section 4.1 of this BAT form for discussion on preference for disposal of corrosion products as solid wastes. **7.1 Solid wastes** The main solid wastes produced during operation of the UK EPR unit, with respect to abatement of Co-58 and Co-60, are Ion Exchange Resins (IER) from the nuclear island and water filters from effluent treatment. According to operating experience feedback at French and German PWRs, and assuming a significant reduction in the volume of operational waste produced, estimated volumes of raw solid wastes are: • 3 $m^3 y^{-1}$ of IER; • 5 $m^3 y^{-1}$ of water filters. It should be noted that the volume of solid waste generated depends on the balance between environmental discharges and packaged waste generation in managing the installation and may therefore change according to the effluent treatment methods. The proposed EPR reference waste management strategy is standardised as closely as possible with that of the Flamanville 3 reactor, including the packaging used (concrete containers, metallic and plastic drums, metallic boxes), whilst recognising the UK specific waste classifications and disposal routes. Ion Exchange Resins from the nuclear island will be treated by embedding and packaged in 2 m^3 concrete blocks, water filters from effluent treatment will be cemented either in 1.23 m^3 to 2 m^3 concrete blocks if ILW, or in 200 litres metallic drums if LLW. Volumes and total activities of solid waste generated are summarised in the table below. It has been conservatively assumed that the physical characteristics of EPR waste arising in operation are essentially the same as the more recent French reactor. In particular, this applies for the prediction of radioactivity content despite the expected reduction in inventory. It should be noted, that treatment and packaging options have been proposed [NESH-G/2008/en/0123/Rev. A - Solid Radioactive Waste Strategy Report] and are currently being evaluated for disposability by NDA-RWMD. Final selection of such alternatives to the reference case will be performed by the future operator.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

Volumes and activities of raw and packaged solid waste (PCER, Sub-chapter 6.3, Table 4)

Type of waste	Estimated gross annual volume (m^3)	Average mass activity (GBq te^{-1} of raw waste)	Estimated gross annual activity (GBq)	Packaging	Number of packages produced per year	Volume of initial packages produced per year (m^3)	Final package for storage or disposal	Volume of final packages produced per year (m^3)	Number of packages to be stored or disposed of per year (upper value)
IER	3	500	1500	2 m^3 concrete blocks	7.5	15	2 m^3 concrete blocks	15.5	8
Water filters (ILW)	2.5	4800	3600	2 m^3 concrete blocks	6	12	2 m^3 concrete blocks	11.4	6
	2.45	1650	1200	1.23 m^3 concrete blocks	9	11.1	1.23 m^3 concrete blocks	12	9
Water filters (LLW)	0.05	< 12	< 0.18	200 litres metallic drums	1	0.2	1/3 height disposal container	0.2	0

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

Co-58 and Co-60 are key radionuclides to consider with respect to the acceptability of solid waste for storage and disposal as they largely contribute to the dose associated with a waste package during the first years of storage. At the time a package is produced, it is estimated that Co-58 and Co-60 contribute respectively 13.6% and 21.1% to the dose arising from packaged IER, and 35.7% and 14.9% respectively to the dose from packaged water filters. Given the relatively short half life of Co-58, its activity becomes negligible after 30 years, and the contribution of Co-60 to the dose from packaged solid waste after a decay period of 30 years would fall to 1.3% for IER and 8% for water filters.

7.2 Liquid Discharges

Co-58 and Co-60 are not expected to have specific liquid discharge limits but are part of the ‘other fission and activation products’ group of radionuclides.

Ionic or particulate forms of corrosion products are usually retained on the filters and demineralisers of the various treatment systems (RCV [CVCS], TEP [CSTS], PTR [FPCS] or 8TEU [LWPS]). However, they still remain detectable in liquid effluents before discharge. According to the average discharges from 1300 MW(e) French PWR units operating over the period 2002 - 2004, Co-58 and Co-60 contribute respectively 20.7% and 30% to the overall activity of fission and activation products discharged in liquid form (PCER, Sub-chapter 11.1, Table 2).

Monthly discharges of these radionuclides are very difficult to predict, as they are very dependent on the effluent management policy adopted on site, the point of the fuel cycle considered and the contingencies encountered. Monthly discharges during shutdown but under normal operating conditions are typically 6 times higher than during operating phases (PCER Sub-chapter 6.3 section 6.4.2).

Overall, considering that the discharges of fission and activation products are not directly linked to the energy produced and, taking into account the design improvements of the EPR compared to the 1300 MW(e) reactors, it is expected that a 10% reduction of the activity discharged in liquid form from the ‘other fission and activation products’ will be achieved compared to the existing PWRs.

BAT FORM FOR COBALT-58 AND COBALT-60 (continued)

7.3 Gaseous Discharges

As for the liquid discharges, Co-58 and Co-60 are not expected to have specific gaseous discharge limits but are part of the ‘other fission and activation products’ group of radionuclides.

The corrosion products remaining in gaseous form are retained in the filtration system of the TEG [GWPS], or in the filtration system of the ventilation systems. According to the average discharges from the 1300 MW(e) French and German PWR units over the period 2002 - 2004, Co-58 and C-60 contribute respectively 25.5% and 30.1% to the overall activity of fission and activation products discharged in gaseous form (PCER, Sub-chapter 11.1, Table 2).

The radioactive discharges from current power stations into the environment as aerosols represent a mass of less than one microgram per year per reactor (largely Co-60). In addition, the discharges from the ventilation systems of the Nuclear Buildings are continuous. These constitute the baseline discharges for fission and activation products, and are generally below limits of detection. The EPR expected performance for the annual discharge of fission and activation products is evaluated at 4 MBq y^{-1}. This very low estimate was based on operating experience feedback from the French 1300 MW(e), and is considered to be equivalent to a sum of detection thresholds.

However, peaks of discharges are also observed, especially during maintenance operations, both during unit shutdown and during operational phases. Moreover, the impact of operational contingencies (e.g. fuel leakage, failure of the treatment systems or of the filters) on the radiochemistry of the primary coolant and hence on gaseous discharges of fission and activation products would be major. Thus, the profile of the fission products and activation products discharges is not expected to be flat over the whole fuel cycle, but prediction of the peaks of discharges are difficult to assess as the profile is completely dependent on the operational contingencies and site operations. In addition, the lack of operating experience feedback for the EPR makes it difficult to predict any unplanned discharges due to maintenance operations. However, it is evident that, due to the very dependent nature of the discharges on these contingencies, the gap between the EPR expected performance and the proposed annual maximum discharge value needs to be significant, as the expected performance without contingencies is equivalent to a record of detection thresholds values. In particular, it is expected that such a situation could lead to a one-off high discharge that could account for over 50% of the total activity discharged over 12 months from releases of gaseous fission and activation products. The maximum gaseous discharge value proposed for activation and fission products is 120 MBq y^{-1}.

3.5 BAT FORM FOR NOBLE GASES

1. Mechanism of production of the radionuclide and how it relates to the process by which the nuclide is produced.	Noble gases are identified, for consideration of PWR gaseous radioactive discharges, as a collection of xenon and krypton radionuclides that are formed by fission in the fuel. According to section 2.7 of Sub-chapter 6.1 of the PCER, the noble gases taken into account in EPR studies are presented in the table below. Xe-133 (half-life of 5.25 days) is the most significant nuclide in terms of activity (see section 2).

Noble Gas	Half-Life
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

Ar-41 is another noble gas produced on a PWR. It is produced by activation of Ar-40 as a minor impurity in the primary coolant. It is not considered here as its contribution to the discharges is relatively insignificant.

Noble gases transfer to the primary coolant as the result of two processes.

BAT FORM FOR NOBLE GASES (continued)

1.1 Fuel leaks

Noble gases can transfer into the reactor coolant by passing through small defects in the fuel cladding. The most common fuel cladding failure causes are grid-to-rod fretting, corrosion and crud, debris, pellet cladding interaction, and manufacturing upsets. Noble gas mobility through these defects is accentuated by changes in reactor conditions, particularly reactor power and coolant pressure, that occur during operations such as reactor shutdown. The sudden increase in noble gas concentrations in reactor coolant that occurs due to reactor power transient is known as fission product spiking.

During normal operation, defective fuel cladding gaps are filled with water, equilibrating with the primary coolant pressure, and the fission product release rate depends on the size of the defect. During power reduction, a portion of the fuel cools down and the primary water is forced into the defect fuel gaps. The decay heat is still high enough to evaporate the water into steam. This process reaches a first spiking peak when the reactor is shut down at zero power. When the reactor is shut down, the pressure in the reactor pressure vessel is maintained at a high level, the fission product release rate is slow and the concentration of fission products in the primary coolant decreases as the let-down system removes the activities. When the pressure starts to drop, the higher pressure inside the fuel cladding begins to push the water - and steam - carrying fission products out of the cladding through the fuel cladding defect. This process reaches a second spiking peak when the pressure drops and the water temperature decreases to near ambient conditions. [Nuclear Science Series NAS-NS-3119 – Radiochemistry in Nuclear Power Reactors, Chien C. Lin, 1996]. Fission product spiking is the first source term of fissions product in the primary coolant.

1.2 “Tramp” uranium

Small quantities of noble gases are produced from uranium contamination deposited in trace amounts on fuel surfaces during the manufacturing process and then deposited in the primary circuit internal surfaces, known as “tramp” uranium.

1.3 Chemical Form

Noble gases are dissolved in the primary coolant depending on the pressure. If the pressure is reduced, a proportion of them will be degassed. They are not present in liquid effluents.

BAT FORM FOR NOBLE GASES (continued)

2. i. Source term (before application of techniques to prevent or minimise at source have been applied) and; ii. Activity in reactor following the application of techniques to prevent or minimise the source term.	U-235 is the source (direct or indirect) of noble gases so, as fission products, their production are related to power output. The majority of the fission products are retained in the fuel by the Zircaloy cladding. The estimated realistic concentrations of noble gases in the primary coolant are presented in the table below. They are based on operating experience feedback from German and French units, and these values are proposed for phases corresponding to steady-state operation and shutdown transient, where fission product spiking occurs (PCER Sub-chapter 6.1 – Tables 1 and 2).

Estimated realistic concentrations of noble gases in the primary coolant of the EPR

Radionuclide	Specific Activity in Stabilised Operation Realistic Value (MBq t^{-1})	Specific Activity at Shutdown Transient Maximum Realistic Value (MBq t^{-1})
Kr-85m	200	460
Kr-85	38	73
Kr-87	360	830
Kr-88	500	1200
Xe-131m	28	53
Xe-133m	110	260
Xe-133	5000	9500
Xe-135	1100	1900
Xe-138	850	2500

BAT FORM FOR NOBLE GASES (continued)

3. Demonstration that the techniques used to minimise at source are BAT by reference to all other options considered and rejected or by benchmarking against best world practice.	The amount of fission products present in the primary coolant is minimised at source by high standards of fuel design and fabrication. The fuel is manufactured to stringent specifications and is subject to rigorous inspection. “Tramp” uranium deposited on fuel surfaces cannot be totally avoided but is only present in trace amounts. Fuel cladding is designed to contain fission products in the fuel as far as possible, but a small number of fuel pins unavoidably have a small number of minute defects through which these fission products can transfer to the primary coolant (known as “failed fuel fraction”). The reactor is operated in such a way as to minimise the risk of fuel failure. As mentioned in section 1.1 of this BAT form, the most common fuel failure causes are grid-to-rod fretting, corrosion and crud, debris, pellet cladding interaction, and manufacturing upsets. An EPRI research programme is dedicated to these issues for which EDF is a programme participant and contributor. EPRI’s Fuel Reliability Action Plan is aimed at understanding the conditions leading to fuel failure and providing a basis for preventing future failures. The programme has a substantial proactive component to establish operating margins under limiting conditions (e.g. changing water chemistries) to ensure fuel operates as specified. In addition, there is a separate programme in the fuels area called Nuclear Fuels Industry Research to conduct basic fuel research. The source term is minimised through several improvements: - minimization of initial surface contamination of the fuel assembly results from the manufacturing process, in particular the clear separation between the “controlled” areas where pellets are manufactured and introduced in the cladding tubes which are then decontaminated before sealing and the “non controlled” areas in which only sealed rods are handled. The surface contamination level is then checked for each fuel assembly; - minimization of crud formation which depends mainly on the control of the primary circuit chemistry and on the operating conditions; - clad failure prevention: o to ensure the fuel rod integrity for all operating conditions, design criteria are defined associated with the temperature of the fuel, the temperature and corrosion of the cladding, the internal pressure / non re-opening of the pellet to cladding gap, the cladding stress, strain and stability, the pellet stack stability, the cladding fretting wear, the cladding fatigue and the fuel rod axial growth;

BAT FORM FOR NOBLE GASES (continued)

	o the design calculations take into account the most relevant uncertainties associated with the manufacturing parameters and the behavior models; o an efficient anti debris device is implemented for fuel assemblies proposed for the EPR; - the annual fuel rod failure rate is a recognized 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 refueled 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 on-going effort to increase fuel reliability whilst reactor operating conditions become increasingly demanding. The fuel performance has been improved in the EPR design, with an anticipated maximum average burn-up in discharged assemblies of 65 GWd per Te, compared to the typical PWR average burn-up of 45 GWd per Te currently achieved (PCER Sub-chapter 8.2 – section 3.2.1). Therefore volumes of spent fuel arisings are much lower than many LWR reactors for each unit of electricity produced. Fuel management options considered in the design phase of the EPR correspond to the optimum of what can be envisaged today using high standard and proven fuel products. For the production of noble gases, minimising the number of fuel elements used minimises the probability for cladding leakage of fission products into the primary coolant. The total quantity of fission products in the reactor is subject to constraints specified in Limiting Conditions of Operation (LCO). These ensure that the concentration of fission products in the primary coolant is controlled in normal operations or in the event of contingencies. In order to minimise the release of fission products caused by fuel leaks, leaking fuel pins are identified during refuelling and not re-used. If there were any major fuel failure at power, significant amounts of noble gases would be released into the primary coolant water; these would be largely degassed in the TEG [GWPS] and treated on delay beds (see section 4.1 of this BAT form). It should be noted that most of the noble gases, with the exception of Kr-85, are short-lived and have no significant radiological impacts during spent fuel management.

BAT FORM FOR NOBLE GASES (continued)

	Overall, the minimisation at source of the fission products present in the primary coolant is considered to be BAT because the EPR uses a proven design of fuel elements with specific power rate similar to those applied in the PWRs, and limits for concentrations ensuring that reactor is not operated with significant defects.
4. Demonstration that the techniques used for treating each process stream minimise the overall impact of the disposal of radioactive waste (including evidence of the consideration of disposal via different pathways).	**4.1 Treatment of Gaseous Effluent** Primary coolant is permanently extracted from the primary circuit by the Chemical and Volume Control System (RCV [CVCS]) to purify, store or degas the primary coolant in different auxiliary systems. All components of these auxiliary systems in which gases can be released from the primary coolant are connected to the TEG [GWPS] system. All these components, tanks etc. are continuously flushed with nitrogen by TEG [GWPS] to process the released gases. For example, during normal operation, a portion of the primary coolant is let-down into the RCV [CVCS] tank. Noble gases are removed from the primary coolant by degassing in the Volume Control Tank (VCT). They then pass into the tank headspace which purges into the TEG [GWPS]. Prior to shutdown of the plant, the TEP [CTS] degasser is started to remove the remaining noble gases to the TEG [GWPS] system. This procedure ensures that there is no remaining gaseous fission product prior to opening of the primary vessel. After plant shut down, prior to opening the reactor pressure head, the reactor pressure head can be flushed with nitrogen by the RPE [NVDS] system to the TEG [GWPS] system. The purpose of the process is to remove spiking noble gases from the primary coolant. The EPR TEG [GWPS] is different to that used on French PWRs but uses the best current methods developed for the German KONVOI design (PCER Sub-chapter 6.2, section 1.2.3). To avoid leaks of gaseous effluents into the Reactor and Auxiliary Buildings, a continuous nitrogen flow runs over the free volumes of the purged tanks and vessels, most of which are kept slightly below atmospheric pressure. The purge gas (nitrogen) after passing through the catalytic recombiner is compressed and then returned to the system for re-use. Thus, a large portion of gas is retained within the TEG [GWPS] system for return to the various ullages and headspaces in the tanks from which it originated, maximising recirculation in a quasi-closed loop and minimising discharges. This retains the shorter-lived noble gases to allow decay prior to discharge to the environment. During plant start-up or plant shutdown only a portion of the purge gas in the system is bled off and fed to dryers to remove water vapour and then to a line of three activated charcoal delay beds.

BAT FORM FOR NOBLE GASES (continued)

	Charcoal adsorption delay beds are used extensively in nuclear reactors around the world and in particular in Sizewell B. Charcoal delay beds offer a more passive system (less moving parts), overall better safety performance and lower operator dose (less pump maintenance). Although delay tanks are another option, they offer no gaseous treatment benefits over charcoal adsorption delay beds and potentially higher operator doses are incurred by their operation and maintenance. The desiccant bed also switches on at reactor start-up and shutdown to dry the gas to ensure maximum efficiency of the delay beds. These retain noble gases that have not already decayed within the recirculating part of the system. They thus provide a further period for the decay of these gases prior to discharge, with xenon isotopes retained on the beds for at least 40 days and krypton isotopes for at least 40 hours (based on German experience with delay beds –PCER Sub-chapter 6.2, section 1.2.3). These timescales are calculated based on the half-lives and the dynamic absorption coefficients of the nuclides (except for Kr-85 which half-life is 10.72 years), the flow rate of the carrier gas and the mass of charcoal; the delay time is between eight and ten times the half-lives of the considered nuclides, as the benefit of further delay is not significant. Considering these minimum retention times, the three delay beds provide minimum reduction factors of 200 and 500 for Xe-133 and Kr-85m respectively. With a half-life of 10.72 years, the delay beds have no significant effect on Kr-85. Concerning short-lived noble gases, the reduction factor is high. The three delay beds also operate at a slightly enhanced pressure to maximise their capacity. A filter is implemented at the outlet of the three delay beds to remove any small carry over of particulates from the beds but this does remove noble gases. After treatment in the TEG [GWPS], the gaseous effluents are directed to the HVAC System (DWN [NABVS]) of the Nuclear Auxiliary Building (NAB) where they are finally treated using one of a series of HEPA filters and iodine traps if needed. Discharge via this route is automatic and controlled via pressure regulation in the TEG [GWPS] delay beds. The final treated gaseous effluents are then monitored and discharged via a stack shared with the HVAC system serving the Nuclear Auxiliary Building. Part of the gaseous fission products associated with the lack of leaktightness of systems carrying primary fluid, are not treated in the TEG [GWPS]. These effluents however pass through the HVAC systems where they are filtered through HEPA filters and iodine traps if needed and are discharged via the stack. If no contingencies occur (e.g. fuel leak, unplanned shutdown), most of the noble gases discharges are expected to occur during the unit shutdown and start-up, when larger volumes of gaseous effluents need to be treated in the TEG [GWPS]. During planned shutdowns, let-down of primary coolant is delayed to allow decay of short-lived noble gases. Discharges are also expected during maintenance operations, when opening of the systems carrying primary coolant is required, although the work can be delayed to allow decay for ALARP purposes.

BAT FORM FOR NOBLE GASES (continued)

The EPR maximum values for discharges of noble gases represent an improvement of 27% as compared to 1300 MW(e) units (per energy produced) (PCER Sub-chapter 8.2 – Table 1). The maximum discharge expected for discharges of noble gases in the EPR also represents nearly half of the limit applied at Sizewell B, normalised to 1000 MW(e) (PCER Sub-chapter 8.2 – Table 2).

4.2 Discussion of the Minimisation of the Radiological Impacts of the Discharge of Noble Gases

In order to consider the minimisation of impacts of radioactive noble gases discharges, it is necessary to consider the radiological impacts in terms of radiation doses to the public. These are discussed in detail in Sub-chapters 6.3 and 11 of the PCER. The methodology for the detailed radiological assessment was based on the use of PC CREAM 98 software tool and is described in Sub-chapter 11.1, section 1.3 of the PCER.

As detailed in section 7.2 of this BAT form, Xe-133 is the main contributor to the total noble gases discharges (63.1%), before Xe-135 (19.8%) and Kr-85 (13.9%) and Xe-131m (0.3%). The contribution of noble gases to the total dose released to air is relatively small, as shown in the table below (PCER Sub-chapter 11, Annex 3 - Table B). It is entirely related to the irradiation from the plume given that noble gases are non-reactive and non-depositing and therefore do not contribute to the inhalation or ingestion pathways. The design of the discharge stack is optimised to minimise the impacts of discharges to atmosphere.

Contribution of noble gases (Xenon and krypton isotopes) to total releases to air

Radionuclide	Contribution to aerial dose (%)
Kr-85	0.015
Xe-133	0.275
Xe-135	0.525
Xe-131m	0.0004
Total	0.8154

BAT FORM FOR NOBLE GASES (continued)

Doses to maximally exposed individuals

The maximum annual discharges of noble gases for an EPR unit were used for the assessment. The tables below set out the calculated doses to the maximally exposed individuals.

Doses to maximally exposed individuals from Annual Releases of noble gases in gaseous effluents

Maximally exposed individuals	Exposure from gaseous discharges (22.5 TBq y^{-1}) ($\mu Sv\ y^{-1}$)
Adult	4.7×10^{-2}
Child	2.9×10^{-2}
Infant	2.3×10^{-2}

The highest total dose calculated to a local resident (adult) from exposure to all radionuclides discharged from an EPR is 21 μSv y^{1}. Although not being one of the major contributors to critical group dose, it is important to keep the limit for the discharges of noble gases ALARP.

Collective Dose

The collective dose is a measure of the radiation exposure in a population and is the sum of effective doses from a given practice or situation to all affected individuals, now and in the future. The environment agencies guidance on prospective doses assessment recognises 500 years as an appropriate measure for collective doses assessments. Collective doses to the populations of UK, Europe and the World, truncated at 500 years, are estimated and presented in PCER Sub-chapter 11.1, section 4. As the site location it is an important factor for estimating the dose to the UK population, it is necessary to make some assumptions about the power station location. Among the potential sites where a new EPR reactor could be located, the site which presents the maximum collective dose is presented in PCER Sub-chapter 11.1, section 4.1. The estimated collective doses are presented in the table below.

BAT FORM FOR NOBLE GASES (continued)

Collective Dose from Annual Releases of noble gases in gaseous effluents

		UK	Europe	World
Collective dose due to noble gases (Man Sv)	**Atmospheric discharges**	8.3×10^{-4}	1.8×10^{-3}	7.9×10^{-9}
Total dose collective (man Sv)		0.31	2.46	16.9

The contributions to the collective dose from discharges of noble gases to UK, European and World populations are negligible.

Potential Short Term Dose

Potential short-term doses have been assessed for gaseous discharges. For the radiological impact assessment of gaseous discharges, it is normally assumed that discharges occur continuously and uniformly over a year. However, during normal operations, short-term gaseous discharges can occur during, for example, reactor shutdown where fission product spiking occurs. It is possible that such short-term discharges may lead to doses that are higher (or lower) than would be expected if it were assumed that the discharges are continuous over a year.

The methodology used and calculated doses are presented in PCER Sub-chapter 11.1, section 2, using the release figures presented in section 2.2 of Sub-chapter 11.1 of the PCER. The table below shows the dose to adults, children and infants from all pathways and radionuclides as a result of a single short term discharge. Despite the conservative assumption used, which is that, all radionuclides are discharged together in the same release; the total short-term dose is less than the total continuous release dose. The contributions to the potential short-term dose from discharges of noble gases to these exposure groups are small, although relatively higher for adults, representing a quarter of the total short-term dose for an adult.

BAT FORM FOR NOBLE GASES (continued)

Potential Short Term Doses

Radionuclide Group	**Effective Dose per discharge (µSv) for a 24 hour scenario**		
	Adult	**Child**	**Infant**
Noble gases	0.033	0.020	0.015
Total short-term dose	0.864	0.889	1.46

5. Demonstration that the techniques used to minimise impacts are BAT by reference to all other options considered and rejected or by benchmarking against best world practice.	Several techniques have been considered for the abatement of discharges of noble gases, which are presented below. The EPR design features an optimised Gaseous Waste Processing System TEG [GWPS] based on charcoal delay beds which is adopted from the German KONVOI design (see section 4.1 of this BAT form). This abates noble gases in gaseous discharges (PCER Sub-chapter 8.2 – section 3.4). In addition, it should be noted that gaseous abatement techniques implemented by the EPR are focused upon technologies considered as best available technologies as mentioned in the OECD Report [Effluent release options from nuclear installations – Technical Background and regulatory aspects, OECD, 2003]. This report presents one principal technique for the abatement of noble gases in gaseous discharges, namely carbon delay beds to retain noble gases, which achieves an economically beneficial retention of the radioactive noble gases. The TEG [GWPS] of the EPR uses the physical surface process of selective dynamic adsorption of noble gases on activated charcoal. This selective dynamic adsorption increases the residence of the noble gases inside the delay beds, allowing them to decay and hence reducing the activity of the carrier gas (nitrogen).The efficiency and normal functioning of the delay beds are monitored by activity measurements downstream of the delay beds. According to German operating experience feedback, failures of delay beds have never occurred on PWRs. The delay beds are designed for 60 years of operation hence the replacement of the charcoal filters is not expected during the life of the plant and the delay beds and generation of solid waste is therefore minimised. Carbon delay beds are considered to be Best Available Technique for abatement of discharges of noble gases and their use on the EPR is considered to be consistent with international best practice.

BAT FORM FOR NOBLE GASES (continued)

	The techniques used in the EPR to minimise impacts of noble gases discharges follow ALARA and ALARP principles, and are comparable to PWRs worldwide and in the UK. Moreover, as mentioned in section 4.1 of this BAT form, the EPR design features optimisation of these techniques and an enhanced treatment system. Overall, the estimations of reductions in discharges discussed in section 4.1 of this BAT form can provide evidence that the EPR complies with a number of BAT principles for the discharge of noble gases. The EPR design presents clear improvements to discharges of noble gases compared to the French PWRs and Sizewell B. In particular, the EPR makes efficient use of resources thereby improving of the eco-efficiency of the facility and reducing discharges. The impact of current discharges of noble gases is so low that the cost of further enhancement of abatement techniques or the development of new techniques would not be justified or cost effective. Operation and maintenance practices, including abatement plant testing and discharge measurements will ensure that abatement remains effective.
6. Demonstration that the techniques used conform to the IWS (cross-reference to IWS sections)	The management of wastes are described in the following sections of the IWS: Section 5 Integrated Waste Management Strategy Section 5.2 Operational radioactive wastes Section 5.2.1 Solid wastes Section 5.2.2 Liquid radioactive wastes Section 5.2.3 Gaseous radioactive wastes. It may be noted that the IWS does not provide detailed discussion on noble gases, which are discussed in detail in this document.
7. Identification of disposals resulting from the use of techniques to minimise overall impacts and quantification of annual volumes and activity of nuclides.	**7.1 Solid Wastes** The main solid waste associated with the abatement of noble gases is the carbon delay bed system so these should be discussed here. However given the short half life of noble gases it would seem likely that the activity in solid waste associated with noble gases would be negligible. The delay beds are designed for 60 years of operation hence the replacement of the charcoal filters is not expected during the life of the plant and the delay beds and generation of solid waste is therefore minimised.

BAT FORM FOR NOBLE GASES (continued)

7.2 Gaseous Discharges

The distribution of the various radionuclides in the discharges of noble gases is presented in the table below. It was determined using the averaged discharges from all current French 1300 MW(e) units and German KONVOI over the period 2002 - 2004. The 1300 MW(e) has been chosen as the reference, since information is readily available, and its design is comparable to the EPR. (PCER Sub-chapter 6.3, section 7.4.2).

Distribution of the activity from noble gases discharges

Radionuclide	**Percentage of total noble gas activity discharged**
Kr-85	13.9
Xe-133	63.1
Xe-135	19.8
Xe-131m	< 1 (0.3)
Total	~ 97

NB: The remaining 3% is associated with Ar-41, which is another noble gas. It is produced by activation of Ar-40 in the air near the core. It is not considered here as its contribution to the discharges is relatively insignificant. The discharges of Ar-41 are related to the start-up of the purging of the Reactor Building (EBA [CSVS]) prior to unit shutdown, and if access is required during operational phases. The discharge then occurs from the Reactor Building ventilation systems rather than from the TEG [GWPS].

BAT FORM FOR NOBLE GASES (continued)

	In normal operating conditions at steady reactor power, it is expected that the EPR monthly discharges of noble gases would be below the limit of detection, and the values recorded for statutory reporting would only represent the sum of detection thresholds. Typically, this could be as low as 400 GBq per month. On the basis of the monthly discharges recorded at Nogent and Golfech and effect of fuel defects, it can be assumed that the noble gas monthly discharges could reach up to 5 TBq per month for the EPR. It has been established that discharges of noble gases over a short period of time can greatly impact the overall discharge over 12 rolling months, as was the case at Golfech in 2005 when 90% of the total activity of the year was discharged in one month. As such, it is essential that a significant headroom remains between the expected performance and the maximum discharge value, for both monthly and annual discharges. It is expected that this maximum value could be reached in case of fuel leaks, at reactor shutdown, and during degassing of the RCV [CVCS] and opening of the primary circuit. (PCER Sub-chapter 6.3, section 7.4.2). The calculation of the EPR expected performance for the noble gases discharges is based on the operating experience feedback data recorded for the first quartile of all the 1300 MW(e) reactors, and the annual maximum discharge value was transposed from the 1300 MW(e) reactors limit (PCER Sub-chapter 6.3, section 7.4.2). The expected performance without contingency and the proposed annual maximum discharge value for the EPR noble gases discharges are thus estimated at 0.8 TBq y^{-1} and 22.5 TBq y^{-1} respectively. The headroom between the EPR expected performance (0.8 TBq y^{-1}) and the proposed annual limit (22.5 TBq y^{-1}) is significant. However, as for the iodine discharges, it has been demonstrated, on the basis of 4 different 1300 MW(e) sites, that the contingencies associated with fuel leaks can have a very large impact on the noble gases discharges (PCER Sub-chapter 6.3, section 7.4.2). Considering the current annual discharge limits in force for the 1300 MW(e) reactors, it is reasonable to keep the same limit for the EPR, which is equivalent to a 25% improvement on a power output equivalent basis. In addition, the impact of noble gases on the total dose received from gaseous emissions is very small and a reduction of the limit would not have a significant impact on reducing overall impact.

3.6 BAT FORM FOR RADIOIODINE ISOTOPES

1. Mechanism of production of the radionuclide and how it relates to the process by which the nuclide is produced.	Halogens are identified, for consideration of PWR liquid and gaseous radioactive discharges, as a collection of iodine radionuclides that are formed by fission in the fuel. According to section 2.7 of Sub-chapter 6.1 of the PCER, the iodine isotopes taken into account in EPR studies are presented in the table below:

Iodine Radionuclide	Half-Life
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

I-131 is the longest lived radionuclide of this category and emits both beta and gamma radiation. As such, it contributes the most to the total iodine dose and is considered to be the most significant of this group in terms of discharges. As all iodine isotopes have a similar chemical behaviour, I-131 is considered to be representative of the iodine isotopes group for the purpose of demonstration of BAT for any evaluation of minimisation and abatement.

Iodine isotopes transfer to the primary coolant as the result of two processes.

1.1 Fuel leaks

Iodine isotopes can transfer into the reactor coolant by passing through small defects in the Zircaloy fuel cladding. The most common fuel cladding failure causes are grid-to-rod fretting, corrosion and crud, debris, pellet cladding interaction, and manufacturing upsets. Iodine isotopes' mobility through these defects is accentuated by changes in reactor conditions, particularly reactor power and coolant pressure that occur during operations such as reactor shutdown. The sudden increase in iodine concentrations in reactor coolant that occurs due to reactor power transient is known as fission product spiking.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

During normal operation, defective fuel cladding gaps are filled with water, equilibrating with the primary coolant pressure, and the fission product release rate depends on the size of the defect. During power reduction, a portion of the fuel cools down and the primary water is forced into the defect fuel gaps. The decay heat is still high enough to evaporate the water into steam. This process reaches a first spiking peak when the reactor is shut down at zero power. When the reactor is shut down, the pressure in the reactor pressure vessel is maintained at high level, the fission product release rate is slow and the concentration of fission products in the primary coolant decreases as the let-down system removes the activities. When the pressure starts to drop, the higher pressure inside the fuel cladding begins to push the water - and steam - carrying fission products out of the cladding through the fuel cladding defect. This process reaches a second spiking peak when the pressure drops and the water temperature decreases to near ambient conditions. [Nuclear Science Series NAS-NS-3119 – Radiochemistry in Nuclear Power Reactors, Chien C. Lin, 1996]. Fission product spiking is the first source term of fissions products in the primary coolant.

1.2 “Tramp” uranium

Small quantities of iodine isotopes are produced from uranium contamination deposited in trace amounts on fuel surfaces during the manufacturing process and then deposited in the primary circuit internal surfaces, known as “tramp” uranium.

1.3 Chemical Form

The chemical form of iodine isotopes in the primary coolant depends on the water chemistry. During operation, where the primary coolant is in alkaline and reducing conditions and the temperature is around 300˚C, only iodide I^- is thermodynamically stable. Calculations for the PWR have shown that the fraction of I_2 and more highly oxidised iodine species (iodate and periodate) and organic species such as iodomethane, should be lower than 10^{-20}. Iodide in the primary coolant is present completely in the ionic dissolved form. [Radiochemistry of Nuclear Power Plants with Light Water Reactors, K-H. Neeb, de Gruyter, 1997].

However, immediately after shutdown, due to boration and addition of H_2O_2, volatile forms of iodine (mainly I_2) have been found in PWRs to vary from a few percent to approximately 40%. A small ratio of iodate is also observed after shutdown, but in some cases, up to 70% of molecular iodine has been reported when H_2O_2 is added to the primary coolant. [Nuclear Science Series NAS-NS-3119 – Radiochemistry in Nuclear Power Reactors, Chien C. Lin, 1996].

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

2. i. Source term (before application of techniques to prevent or minimise at source have been applied) and; ii. Activity in reactor following the application of techniques to prevent or minimise the source term.	U-235 is the source (direct or indirect) of iodine isotopes so, as fission products, their production are related to power output. The majority of the fission products are retained in the fuel by the Zircaloy cladding. The estimated realistic concentrations of iodine isotopes in the primary coolant are presented in the table below. They are based on operating experience feedback from German and French units, and these values are proposed for phases corresponding to steady-state operation and shutdown transient, where fission product spiking occurs (PCER Sub-chapter 6.1 – Tables 1 and 2).

Estimated realistic concentrations of iodine isotopes in the primary coolant of the EPR

Radionuclide	Specific Activity in Stabilised Operation Realistic Value (MBq t^{-1})	Specific Activity at Shutdown Transient Maximum Realistic Value (MBq t^{-1})
I-131	100	2300
I-132	190	2200
I-133	310	2400
I-134	190	1500
I-135	200	1400

3. Demonstration that the techniques used to minimise at source are BAT by reference to all other options considered and rejected or by benchmarking against best world practice.	The amount of fission products present in the primary coolant is minimised at source by high standards of fuel design and fabrication. The fuel is manufactured to stringent specifications and is subject to rigorous inspection. “Tramp” uranium deposited on fuel surfaces cannot be totally avoided but is only present in trace amounts. Fuel cladding is designed to contain fission products in the fuel as far as possible, but a small number of fuel pins unavoidably have a small number of minute defects through which these fission products can transfer to the primary coolant (known as “failed fuel fraction”).

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

	The reactor is operated in such a way as to minimise the risk of fuel failure. As mentioned in section 1.1 of this BAT form, the most common fuel failure causes are grid-to-rod fretting, corrosion and crud, debris, pellet cladding interaction, and manufacturing upsets. An EPRI research programme is dedicated to these issues for which EDF is a programme participant and contributor. EPRI's Fuel Reliability Action Plan is aimed at understanding the conditions leading to fuel failure and providing a basis for preventing future failures. The programme has a substantial proactive component to establish operating margins under limiting conditions (e.g. changing water chemistries) to ensure fuel operates as specified. In addition, there is a separate programme in the fuels area called Nuclear Fuels Industry Research to conduct basic fuel research. The fuel performance has been improved in the EPR design, with an anticipated maximum average burn-up in discharged assemblies of 65 GWd per Te, compared to the typical PWR average burn-up of 45 GWd per Te currently achieved (PCER Sub-chapter 8.2 – section 3.2.1). Therefore volumes of spent fuel arisings are much lower than many LWR reactors for each unit of electricity produced. Fuel management options considered in the design phase of the EPR correspond to the optimum of what can be envisaged today using high standard and proven fuel products. For the production of iodine, minimizing the number of fuel elements used minimises the probability for cladding leakage of fission products into the primary coolant. The total quantity of fission products in the reactor is subject to constraints specified in Limiting Conditions of Operation (LCO). These ensure that the concentration of fission products in the primary coolant is controlled in normal operations or in the event of contingencies to ensure that worker and public doses are ALARP. In order to minimise the release of fission products caused by fuel leaks, leaking fuel pins are specifically located during refuelling and not reused. If there were any major fuel failure at power, which is considered to be a highly unlikely occurrence, significant amounts of iodine isotopes would be released into the primary coolant water and iodine adsorption beds on gas discharge outlets would be put in service (see section 4.2 of this BAT form) to minimise any discharges required to recover from the event and return the reactor to power. Overall, the minimisation at source of the fission products present in the primary coolant is considered to be BAT because the EPR uses a proven design of fuel elements with specific power rate similar to those applied at other PWRs, and limits for concentrations ensuring that reactor is not operated with significant fuel cladding defects.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

4. Demonstration that the techniques used for treating each process stream minimise the overall impact of the disposal of radioactive waste (including evidence of the consideration of disposal via different pathways).	**4.1 Treatment of Liquid Effluent** During normal operation, a portion of the primary coolant is let-down into the RCV [CVCS] tank. Fission products are removed from the primary coolant using the ion exchange resins in the RCV [CVCS] with volatile products being removed in the Volume Control Tank (VCT). They then pass into the tank headspace which purges into the TEG [GWPS]. Because iodine isotopes are mainly present in the form of iodide during normal operation, they will be treated on filters and demineralisers with the liquid effluents. Since iodine isotopes are well retained by the primary coolant and liquid radwaste demineralisers and specific activity is low, iodine should not be normally detected in significant quantities in liquid discharges (PCER Sub-chapter 6.3, section 6.4.1.1). General iodine isotopes discharges usually happen at shutdown and start-up, and during maintenance operations requiring the opening of the RCP [RCS]. The purification media of the RCV [CVCS] have been optimised in the EPR design in order to increase the life and efficiency of ion exchange resins and filters. Although this optimisation is focussed on the abatement of the more radiologically significant radionuclides rather than iodine isotopes [ECEF072084 – EPR Flamanville 3 - Summary report: specific design provisions related to EPR chemistry, February 2008]. Primary liquid effluents are then sent to the TEP [CSTS] where they can be treated by evaporation, filtration and demineralisation. The use of degasser in the TEP [CSTS] to ensure that fission products are transferred from the primary coolant prior to opening the reactor vessel is outlined in section 4.2 below. In any case, the small amount of iodine present in the primary coolant and not retained by the demineralisation would still transfer into liquid effluent discharges. The non-recycled liquid effluents arising from the TEP [CSTS] are then treated in the 8TEU [LWPS] and collected into the on-site storage tanks before monitoring and discharge. The effluents from the Process Drain (PD), the Chemical Drain (CD) and the Floor Drains (FD1 and FD2 only) are also treated in the 8TEU [LWPS]. (PCER Sub-chapter 6.2, section 1.1.2). As mentioned above, iodine isotopes are preferentially transferred to the liquid phase during shutdown and restart phases due to their chemical form as iodides, which are highly soluble and not readily degassed. If the reactor is restarted and returned to full power quickly it is necessary to process and dispose of iodine isotopes promptly to allow dilution to criticality, which would not be achieved through the gaseous discharge pathway.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

As no major design changes have been made with the intent of reducing overall iodine liquid discharges from the EPR compared to a 1300 MW(e) unit, it is expected that the discharge values would remain similar for both installations. Although the EPR absolute values (expected performance and maximum discharges) are similar to those expected for a 1300 MW(e) unit, they represent a 25% reduction in real terms when they are compared to the energy produced by both units (PCER Sub-chapter 8.2 – Table 1). The maximum discharge expected for liquid discharges of iodine isotopes is also 18 times lower for the EPR than those reported for Sizewell B, normalised to 1000 MW(e) (PCER Sub-chapter 8.2 – Table 2).

4.2 Treatment of Gaseous Effluent

In order to reduce the concentration of gaseous fission products dissolved in the RCP [RCS], the EPR design comprises two degassing processes [ECEF072084 – “EPR Flamanville 3 - Note de synthèse : dispositions particulières de conception liées à la chimie de l’EPR *(Summary note: design specificities linked to the EPR chemistry)*”, February 2008]:

- a degasser on the outlet of the RCV [CVCS], which is part of the TEP [CSTS] and linked to the TEG [GWPS];
- two deconcentration lines for the vapour phase of the pressuriser, linked to the Nuclear Vent and Drain System (RPE [NVDS]).

The iodine isotopes degassed in the TEP [CSTS] are then sent to the TEG [GWPS]. Also, the gaseous iodine isotopes present in the RCV [CVCS] tank pass into the tank headspace which purges into the TEG [GWPS]. The EPR TEG [GWPS] is different to that used on French PWRs but uses the best current methods developed for the German KONVOI design, summarised below and considered by benchmarking as the preferred approach (PCER Sub-chapter 6.2, section 1.2.3.1).

To avoid leaks of gaseous effluents into the Reactor and Auxiliary Buildings, a continuous nitrogen flow runs over the free volumes of the purged tanks and vessels, most of which are kept below atmospheric pressure. The recovery of purge gas (nitrogen) after passing through the catalytic recombiner is compressed and then returned to the system for re-use. Thus, a large portion of gas is retained within the TEG [GWPS] system for return to the various ullages and headspaces in the tanks from which it originated, maximising recirculation and minimising discharges. This retains the shorter-lived radioactive gases to allow decay, including short lived iodine species, prior to discharge to the environment.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

Hydrogen and oxygen levels lost to the circulating gas from the liquids in the tanks to the purge gas are controlled using a catalytic recombiner which recombines the hydrogen and oxygen to steam and condensed to water. In the TEG [GWPS] waste gas compressor unit an intensive gas / water contact takes place. This is caused by the functional principle of the liquid ring compressor. The TEG [GWPS] recombination unit and the TEG [GWPS] waste gas compressor unit may help to ensure that iodine isotopes in the purge gas in the TEG [GWPS] are returned to and retained in the liquid phase.

According to the “European Utility Requirements for LWR Nuclear Power Plants Section 2.1A-13” the chemical form of iodine is defined as:

- Iodine, particulate 95.00% (mostly caesium iodide);
- Iodine, elemental and as HI 4.85%;
- Iodine, organic 0.15%.

The particulate and elemental forms of iodine are either solid or liquid, only HI is gaseous. Depending upon their composition, the organic forms of iodine have different boiling points with methyl iodide (CH_3I) having the lowest (42.4°C). Therefore effective removal of iodine from the gas in the GWPS at low temperature and condensing conditions is possible.

A portion of the purge gas in the system is bled off and fed to dryers (desiccant beds) to remove water vapour and then to a line of three activated carbon delay beds. The desiccant bed is also activated at reactor start-up and shutdown to dry the gas and ensure maximum efficiency of the delay beds. Although these beds are not specifically targeted for the abatement of iodine isotopes, which are retained mainly in the liquid phases in the TEG system (where the water/gas come into contact in the TEG [GWPS] waste gas compressor; see above), the delay beds have the same decay effect as for the abatement of xenon (40 days hold-up) due to the mass of iodine molecules and the adsorption capabilities of the charcoal (PCER Sub-chapter 6.2, section 1.2.3.1). The three delay beds operate at a slightly enhanced pressure to maximise their capacity. They finally feed through to a filter to remove any small carry over of particulates from the bed.

After treatment in the TEG [GWPS], the gaseous effluents are directed to the HVAC system (DWN [NABVS]) of the Nuclear Auxiliary Building (NAB) where they are finally treated using one of a series of HEPA filters and iodine traps when necessary. Discharge via this route is automatic and controlled via pressure in the TEG [GWPS] delay beds. The final treated gaseous effluents are then monitored and discharged via a stack shared with the HVAC systems.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

	Part of the gaseous iodine isotopes associated with the lack of leak tightness of systems carrying primary fluid, are not treated in the TEG [GWPS]. These effluents however pass through the HVAC systems which incorporate High Efficiency Particulate Air (HEPA) filters and can be switched to iodine traps in the event of a significant release of iodine isotopes. The switch of the ventilation to iodine traps can be implemented either automatically if a high activity has been detected by the KRT [RPMS] system or as a preventive method during certain operations or in case of an accident for the Safeguard and Reactor Buildings. As an example, the detection level in the French NPPs is set to around one-tenth of the regulatory limit. The decontamination factor of the traps is 100 for systems which operation is required during fault conditions and 10 for the others; these decontamination factors are set up in accordance with studies of radiological consequences (see PCER Sub-chapter 6.2 – section 1.2.3.2.1 for more details on the iodine traps). Sampling of iodine isotopes in the stack will be carried out by transferring a portion of the gas taken for particulate sampling and passing it through charcoal that has been specially treated to make it adsorb halogens very efficiently. The halogens adsorption technique used for sampling is the same as that used for reduction of iodine isotopes in gaseous discharges. Each discharge outlet charcoal sample is analysed for radioactive iodine isotopes using commercially available gamma spectrometry counting equipment (PCER Sub-chapter 8.4 – section 6.1.2, the KRT measurements which ensure control in the ventilation system and switch the system to use of iodine traps are based on gross beta counting). If no contingencies occur (e.g. fuel leak, unplanned shutdown), most of the gaseous iodine discharges are expected to occur during the unit shutdown and start-up, when larger volumes of gaseous effluents need to be treated in the TEG [GWPS]. Discharges are also expected during maintenance operations, when opening of the systems carrying primary coolant is required. The EPR values for the discharge of gaseous iodine isotopes represent an improvement of 20% as compared to 1300 MW(e) units on a power output equivalent basis, and 25% over the maximum discharge value per MW(e) produced by a 1300 MW(e) unit (PCER Sub-chapter 8.2 – section 4.2.2 and Table 1). The maximum discharge expected for gaseous discharges of iodine isotopes in the EPR also represents nearly half of the limit applied at Sizewell B, normalised to 1000 MW(e) (PCER Sub-chapter 8.2 – Table 2).

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

4.3 Discussion of the Minimisation of the Radiological Impacts of the Discharges of Iodine isotopes

Further optimisation for minimising radioactive discharges impacts will be achieved by consideration of stack design and liquid effluent discharge arrangements during the site specific evaluations.

In order to consider the minimisation of impacts of radioactive iodine isotopes, it is necessary to consider the radiological impacts in terms of radiation doses to the public. These are discussed in detail in Sub-chapters 6.3 and 11 of the PCER. The methodology for the detailed radiological assessment was based on the use of PC CREAM 98 software tool and is described in Sub-chapter 11.1, section 1.3 of the PCER.

I-131 and I-133 contribute respectively 45.6% and 54.4% to the maximum gaseous iodine discharges (PCER Sub-chapter 11.1, Table 2). According to Tables B,C et D et E, F, G Appendix 3 of the PCER Chapter 11:

- For gaseous discharges, I-131 is the main contributor to the dose due to iodine isotopes in gaseous (95% - 98%), and is responsible for 1% up to 4% of the total dose in gaseous discharges.
- For liquid discharges, I-131 is considered to be the only contributor to the liquid iodine discharges and its contribution to the total liquid discharges is negligible (< 0.0015%).

Doses to maximally exposed individuals

The maximum annual liquid and gaseous discharges for an EPR unit were used for the assessment. The tables below set out the calculated doses to the maximally exposed individuals from both liquid and gaseous discharges.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

Doses to maximally exposed individuals from Annual Releases of iodine isotopes in gaseous and liquid effluents

Maximally exposed individuals	Exposure from gaseous discharges (400 MBq y^{-1}) ($\mu Sv\ y^{-1}$)	Exposure from liquid discharges (50 MBq y^{-1}) ($\mu Sv\ y^{-1}$)	Local resident exposure from both liquid and gaseous discharges ($\mu Sv\ y^{-1}$)
Adult	3.9×10^{-2}	7.6×10^{-5}	3.9×10^{-2}
Child	7.8×10^{-2}	3.8×10^{-5}	7.8×10^{-2}
Infant	3.2×10^{-1}	2.2×10^{-5}	3.2×10^{-1}

It is theoretically possible that a family (adults, children and infants) living in the nearest habitation to be exposed to both atmospheric discharges and to liquid discharges in the marine environment. To determine the dose to these local residents it is necessary to sum the doses from aerial discharges and liquid discharges, and these are shown in the above table for completeness, although the contribution from iodine liquid discharges is negligible compared to the contribution from gaseous discharges. The highest total dose calculated to a local resident (adult) from exposure to all radionuclides discharged from an EPR is 21 μSv y^{-1}. As such, the discharges of liquid iodine isotopes would only have a minor radiological impact on the overall dose received by the most exposed members of the public. Although not being one of the major contributors to critical group dose, it is still necessary for the discharges of gaseous iodine isotopes to be ALARP.

Collective Dose

The collective dose is a measure of the radiation exposure in a population and is the sum of effective doses from a given practice or situation to all affected individuals, now and in the future. The environment agencies guidance on prospective doses assessment recognises 500 years as an appropriate measure for collective doses assessments. Collective doses to the populations of UK, Europe and the World, truncated at 500 years, are estimated and presented in PCER Sub-chapter 11.1, section 4. As the site location is an important factor for estimating the dose to the UK population, it is necessary to make some assumptions about the power station location.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

Among the potential sites where a new EPR reactor could be located, the site which presents the maximum collective dose is presented in PCER Sub-chapter 11.1, section 4.1. The estimated collective doses are presented in the table below.

Collective Dose from Annual Releases of iodine isotopes in gaseous and liquid effluents

		UK	Europe	World
Collective dose (Man Sv)	**Atmospheric discharges**	1.1×10^{-4}	1.2×10^{-4}	N/A
	Liquid discharges	1.8×10^{-8}	2.3×10^{-8}	2.3×10^{-8}
	Total	1.1×10^{-4}	1.2×10^{-4}	2.3×10^{-8}
Total collective dose (man Sv)		0.31	2.46	16.9

The contributions to the collective dose from discharges of iodine isotopes to UK, European and World populations are negligible.

Potential Short Term Dose

Potential short-term doses have been assessed for gaseous discharges. For the radiological impact assessment of gaseous discharges, it is normally assumed that discharges occur continuously and uniformly over a year. However, during normal operations, short-term gaseous discharges can occur during, for example, reactor shutdown where fission product spiking occurs. It is possible that such short-term discharges may lead to doses that are higher (or lower) than would be expected if it were assumed that the discharges are continuous over a year.

The methodology used and calculated doses are presented in PCER Sub-chapter 11.1, section 2, using the release figures presented in section 2.2 of Sub-chapter 11.1 of the PCER. The table below shows the dose to adults, children and infants from all pathways and radionuclides as a result of a single short term discharge. Despite the conservative assumption used, which is that, all radionuclides are discharged together in the same release; the total short-term dose is less than the total continuous release dose. The contributions to the potential short-term dose from gaseous discharges of iodine isotopes to these exposure groups are very small.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

	Potential Short Term Doses
5. Demonstration that the techniques used to minimise impacts are BAT by reference to all other options considered and rejected or by benchmarking against best world practice.	Several techniques have been considered and implemented for the abatement of discharges of iodine isotopes, which are presented below.

Potential Short Term Doses

Radionuclide Group	Effective Dose per discharge (µSv) for a 24 hour scenario		
	Adult	Child	Infant
Iodine isotopes	0.009	0.011	0.023
Total short-term dose	0.864	0.889	1.46

5. Demonstration that the techniques used to minimise impacts are BAT by reference to all other options considered and rejected or by benchmarking against best world practice.

Several techniques have been considered and implemented for the abatement of discharges of iodine isotopes, which are presented below.

5.1 Liquid Discharges

Demineralisers

It should be noted that the selection of the most appropriate ion-exchange media is based on the abatement of a range of radionuclides for which iodine is not a principal radionuclide. However it is removed by the use of the demineralisers. The demineralising operations involve passing the effluent through resin beds that fix the elements present in ionic form in the effluent. The filtration-decontamination process used in the TEP [CSTS] system comprises a mixed-bed demineraliser containing resins that reduces activity, including abatement of radioactive iodine isotopes in the primary effluent allowing them to decay. Demineralisers are also used in the 8TEU [LWPS] to reduce activity of the spent effluents arising from the TEP [CSTS] and drains (process drains and active floor and chemical drains). The resin beds used are strong high-capacity anionic or macro-porous resins, and strong high-capacity gel-type cationic resins, to optimise removal of ions from the effluents (PCER Sub-chapter 8.2 – section 3.3.3).

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

The initial choice retained for the UK EPR is one high-capacity cationic bed, one strong high-capacity gel-type cationic bed and one mixed bed. Each bed can be by-passed, therefore the effluents can be routed via the demineralisers in any order. This enables operator flexibility, for example if there is a problem with one of the beds or for maintenance to be carried out on a bed, without interruption of the demineralisation process. It also allows flexibility in dealing with specific pollutants (in particular silver), as one bed can be used for a specific treatment if necessary. Moreover, in order to further reduce radioactive discharges, effluent can be recirculated in the beds (one or more beds subject to sampling controls results) until they are acceptable for discharge to the storage tanks (T (0KER [LRMDS]) or S (0TER [ExLWDS]) tanks).

The minimal and maximal volumes of the demineralisers are calculated based on the velocity, the loading per hour and the dimensions of the bed. The quantity of resin per bed is then optimised between the space left for maintenance and the frequency of the replacement. The volume determined for the Flamanville 3 EPR is 2.1 m^3 and the replacement frequencies are approximately 1.5 years for the cationic bed and yearly for the mixed bed. Using the maximum volume would not bring about a significant gain in the reduction of the frequencies.

There are two successive stages in the demineralisation operation. First, effluents are recirculated. Then, the demineraliser operates in open circuit, when the activity has been controlled and the treated effluents are sent to the storage tanks (T (0KER [LRMDS]) or S (0TER [ExLWDS]) tanks) for monitoring and discharge. The demineralisation operations and the use of ion-exchange resins enable the minimisation of radioactive waste generated by concentration of the activity, and the creation of waste into a manageable solid form (sorbed onto the resins).

The French OPEX on the liquid discharges of iodine isotopes (PCER Sub-chapter 6.3 – section 6.4.1) indicate that the use of demineralisers is a proven and very effective abatement technique.

Delay and monitoring tanks

Within the EPR concept, any discharge of liquid radioactive effluent into the environment is subject to prior measurement of volume and activity in the storage tanks upstream of the discharge point. This measurement is performed in the laboratory after a representative sample of the effluent to be discharged is taken, the result of which determines whether the effluent may be discharged into the environment or if it requires further treatment or decay storage. On the discharge pipeline of the on-site storage tanks, a volume activity monitor control is installed. This activates an alarm in the control room and causes the automatic closure of a valve which stops the discharge in case a pre-set threshold is exceeded. The discharge pipeline of the 0KER [LRMDS] tanks is also automatically isolated on a flow rate variance (PCER Sub-chapter 8.4 – section 3.2).

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

5.2 Gaseous Discharges

The EPR design features the following improvements to abate iodine isotopes in gaseous discharges (PCER Sub-chapter 8.2 – section 3.4):

- an optimised Gaseous Waste Processing System TEG [GWPS] adopted from the German KONVOI design (see section 4.2 of this BAT form);
- an improved Ventilation/Filtering system – all of the ventilation systems for the Nuclear Auxiliary Building (NAB), Safeguard Building (SB), Fuel Building (FB) and the Effluent Treatment Building (ETB) controlled areas can be routed to iodine traps prior to discharge if necessary. This is in contrast to the current 1300 MW(e) plants where only selected plant areas in the NAB can be so routed.

In addition, it should be noted that gaseous abatement techniques implemented by the EPR are focused upon technologies considered as best available technologies as mentioned in the OECD Report [Effluent release options from nuclear installations – Technical Background and regulatory aspects, OECD, 2003].

This report includes the use of carbon delay beds to remove gaseous species such as iodine isotopes and noble gases, which are implemented on the EPR (see section 4.2 of this BAT form).

5.3 Conclusions

The techniques used in the EPR to minimise impacts of iodine isotopes discharges follow ALARA and ALARP principles, and are comparable to, or are in some cases better than PWRs worldwide and Sizewell B. Moreover, as mentioned in sections 4.1 and 4.2 of this BAT form, the EPR design features optimisation of these techniques and enhanced treatment systems.

Overall, the estimations of reductions in discharges discussed in sections 4.1 and 4.2 of this BAT form provides evidence that BAT is being applied with respect to the discharge of iodine isotopes. The EPR design presents clear improvements to the liquid and gaseous discharges of iodine isotopes compared to the French PWRs and Sizewell B. In particular, the EPR makes efficient use of resources thereby improving the eco-efficiency of the facility and reducing discharges.

The impact of current discharges of iodine isotopes from existing reactors is so low that the cost of further enhancement of abatement techniques or the development of new techniques for the EPR would not be justified or cost effective. Operation and maintenance practices, including abatement plant testing and discharge measurements will ensure that abatement remains effective.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

6. Demonstration that the techniques used conform to the IWS (cross-reference to IWS sections)	The management of wastes are described in the following sections of the IWS: Section 5 Integrated Waste Management Strategy Section 5.2 Operational radioactive wastes Section 5.2.1 Solid wastes Section 5.2.2 Liquid radioactive wastes Section 5.2.3 Gaseous radioactive wastes. It may be noted that the IWS does not provide detailed discussion on isotopes of iodine, which is discussed in detail in this document.
7. Identification of disposals resulting from the use of techniques to minimise overall impacts and quantification of annual volumes and activity of nuclides.	**7.1 Solid Wastes** Ion exchange resins from the demineralisers are the main solid wastes generated by abatement of iodine isotopes. Details are available in Sub-chapter 6.3 – section 3 of the PCER. However, given the relatively short half-lives of the iodine radionuclides its disposal is not of significant interest for the management of spent ion-exchange resins. Data provided to NDA/RWMD and to LLWR for the purposes of the GDA disposability assessment include activities of specific nuclides in solid waste. With regard to operational VLLW and LLW, LLWR technical experts have performed their review of the information provided on the D1 forms and have provided a letter confirming in principle the acceptability of the UK EPR LLW waste streams for treatment / disposal through LLWR [Form D1 Application: UK EPR Project, from LLW Repository Ltd. 320.L.027. December 2008]. The assessment of operational ILW management is still in progress. **7.2 Liquid Discharges** Operating experience feedback from French PWRs shows that some variations in monthly I-131 liquid discharges, reflecting the fact that the profile of these discharges over a whole fuel cycle will fluctuate significantly. Although it has been established that the I-131 discharges are higher during reactor shutdown and at start-up, this is not necessarily obvious on the basis of review of discharge levels.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

Operating experience feedback based on the first quartile of the entire fleet of the 1300 MW(e) reactors in France has shown liquid iodine discharges around 7 MBq y^{-1}. This very low value is equivalent to a sum of detection thresholds, and considering the EPR specificities, is used as the EPR annual expected performance, in the absence of contingency (PCER Sub-chapter 6.3, section 6.4.1).

However, higher values are expected due to potential contingencies. Major contingencies have been identified as being related to fuel leaks and faults of treatment systems, which are likely to impact liquid discharges of iodine isotopes. It is not expected that the occurrence of a single contingency would greatly affect the discharge. Indeed, it is evident from the OEF from French PWRs that the liquid discharges of iodine isotopes are usually low and account for a sum of detection threshold values, rather than actual measured data. Although the concentration of liquid iodine isotopes can sometimes be high in the primary circuit, the confinement of the systems and the treatment systems of the effluents are usually very efficient in removing these radionuclides prior to discharge.

Nevertheless, a combination of two or more contingencies, including failure of one or more of the treatment systems could have a major impact on the discharges of liquid iodine isotopes. In that situation, the iodine isotopes discharges would be limited over that time period but could reach very high values. Indeed, a worst case scenario involving fuel leaks, unavailability of the treatment systems and the unavailability of one of the delay tanks would mean that the liquid effluent contaminated with iodine isotopes would need to be discharged promptly. In this scenario, an assumption is that the contingency could contribute for up to 90% of the headroom provided on the annual estimates, and a maximum monthly value of 50 MBq per month could potentially be discharged. Although this high discharge would be very limited in terms of the time period, the fact that discharge values in normal operating conditions are equivalent to the sum of threshold values (and therefore very low) means that this would greatly affect the annual total discharge value. These issues support the need for headroom, in order to allow for such operational contingencies. The EPR maximum annual discharge for liquid iodine isotopes (including I-131) is estimated at 50 MBq y^{-1}. This is consistent with the higher values of 40 MBq y^{-1} and 130 MBq y^{-1} observed in France, respectively at Nogent (2 units) and Paluel (4 units) in 2002. (PCER Sub-chapter 6.3, section 6.4.1).

7.3 Gaseous Discharges

The design of the EPR gaseous treatment system is different to that implemented on the existing 1300 MW(e) reactors (see section 4.2 of this BAT form), and, as such, comparison between the two may only be partially relevant. In addition, operational conditions that can affect such discharges will be different in the EPR and in the 1300 MW(e) reactors.

BAT FORM FOR RADIOIODINE ISOTOPES (continued)

However, OEF of the 1300 MW(e) reactors can provide some useful information. In particular, fuel failures were the source of relatively high discharges reaching between 100 and 200 MBq y^{-1} per unit, at Flamanville, Paluel, Golfech and Nogent, although most annual iodine discharges are below 50 MBq y^{-1}per unit at the first three sites (PCER Sub-chapter 6.3, section 7.4.3). These values illustrate that the gaseous iodine discharges are very dependent on fuel failures.

Monthly operational data from the 1300 MW(e) French units between 2002 and 2007 show that the high annual discharges seen for some of the years and at some of the sites are usually only due to a small number of high monthly discharges in the year, rather than an overall higher baseline. Although most monthly discharges over the whole period did not exceed 5 MBq per month, peaks were observed at Flamanville 1, Paluel and Golfech, representing respectively 40%, 60% and 90% of the total activity discharged over 12 rolling months from gaseous iodine isotopes. Again, theses examples illustrate the very high dependency of gaseous iodine discharges on fuel failures, and the fact that discharges limited in time can greatly impact the annual activity discharged from these radionuclides.

Considering the operating experience feedback mentioned above and the EPR expected performance and maximum discharge values, it can be expected that, in a scenario without any fuel failures, the monthly gaseous iodine discharges would be in the order of 20 MBq. For circumstances involving fuel leaks, shutdown and failure of the gaseous effluent treatment systems, it is expected that this value could reach up to 300 MBq per month, equivalent to 75% of the proposed annual maximum discharge value (PCER Sub-chapter 6.3, section 7.4.3). These issues support the need for significant headroom between the EPR expected performance and the proposed annual maximum discharge value.

Operating experience feedback based on the first quartile of the entire fleet of the 1300 MW(e) reactors in France has shown low values for gaseous iodine discharges of around 50 MBq y^{-1}. This very low value is equivalent to a sum of detection thresholds as opposed to actual measured discharge values. In addition, the very low limit values set for the 1300 MW(e) reactors (400 MBq y^{-1}) do not offer significant headroom for contingencies, as shown in the examples mentioned above. To determine the EPR expected performance, it is realistic to incorporate a gain taking into consideration the improvements made to the TEG [GWPS] system, and to use the 1300 MW(e) limit for the maximum discharge value. As such the annual discharge value of 50 MBq y^{-1} is used as the EPR annual expected performance, in the absence of contingency, and the EPR maximum discharge value is expected to be as low as 400 MBq y^{-1} (PCER Sub-chapter 6.3, section 7.4.3).

4. PROPOSALS FOR GROUPING OF SIGNIFICANT RADIONUCLIDES FOR LIMIT SETTING

This section provides information on proposals for:

- the identification of radionuclides that it is proposed should be individually limited;
- the identification of radionuclides it is proposed should be grouped for the setting of disposal limits.

The proposed groupings of radionuclides for the setting of disposal limits for gaseous and liquid radioactive discharges are listed below, together with information on discharges in terms of both annual expected performance excluding contingency and maximum annual discharges taking account of the need for contingencies. Further detailed justification of the selection of the radionuclides and the estimated discharges is provided in Sub-chapter 6.3 of the PCER and will be provided as part of a site-specific submission under RSA'93. The principal reasons for their inclusion and grouping, where relevant, are that they have been identified as significant in the assessment (section 2) and can be measured for monitoring purposes using proven techniques.

Gaseous discharges

Category	Annual expected performance excluding contingency (GBq)	Maximum annual gaseous radioactive discharges (GBq)
Tritium	500	3000
Carbon-14	350	700
Iodine isotopes	0.05	0.4
Noble gases	800	22,500
Other fission or activation products emitting beta or beta and gamma radiation (Excluding Tritium, Iodine or Carbon-14)	0.004	0.120

Liquid discharges

Radionuclide or Group of Radionuclides	Expected annual performance (excluding contingency) for liquid radioactive discharges (GBq)	Maximum annual liquid radioactive discharges (GBq)
Tritium	52,000	75,000
Carbon-14	23	95
Iodine isotopes	0.007	0.05
Other fission or activation products emitting beta or beta and gamma radiation (excluding Tritium, Iodine or Carbon-14)	0.6	10

It is thus proposed that two of the three radionuclides of high significance, carbon-14 and tritium should be individually limited for both gaseous and liquid discharges. However, it is proposed that the third significant radionuclide group, isotopes of cobalt (cobalt-60 and cobalt-58) be grouped with other fission or activation products, as it is difficult to separately measure and predict cobalt-60 and cobalt-58 individually and the grouped radionuclides are not discharged in significant quantities sufficient to warrant separate reporting. It is proposed that iodine isotopes are grouped for the purpose of liquid and gaseous discharges and that noble gases are grouped for gaseous discharges.

At present no specific limits are sought for the grouping of radionuclides for the purpose of disposing of solid wastes. However, proposals for the disposal of solid wastes will comply with the conditions for acceptance for each disposal route and nuclide groups, and limits will be specified as required.

5. CONCLUSIONS

The information presented in this report demonstrates that the Best Available Techniques (BAT) are being applied in the design of the EPR to minimise radioactive wastes at source and to minimise the impacts of the disposal of wastes into the environment. A number of conclusions can be drawn from the assessment presented:

- Most radionuclides arising from the operation of the EPR have been assessed as being of low significance in terms of their radiological impacts and the magnitude of discharges. Summary information has been presented on how radionuclides are minimised at source and the impacts of their disposal are minimised;

- Only a small number of radionuclides have been identified as having high or medium significance (carbon-14, tritium, isotopes of cobalt, iodine isotopes and noble gases) and thus require further assessment with respect to the demonstration of BAT.

- Carbon-14 is minimised at source by measures aimed at increasing the efficiency of nuclear fuel which reduces its production per unit of fuel. The majority of carbon-14 is discharged to the environment in gaseous form, with only relatively small proportions being discharged in liquid discharges or solid wastes. Discharge in gaseous form has a lower radiological impact, per unit discharged, than discharge in liquid form. Discharge of the majority in gaseous form is therefore considered to be BAT;

- No practicable abatement techniques for the removal of carbon-14 from liquid and gaseous discharges have been identified, other than the removal of carbon-14 from liquid discharges which occurs as a result of primary coolant treatment processes which are aimed at removal of other species for operational and nuclear safety purposes. The proposed operation of the EPR is consistent with international best practice for PWRs with respect to the management of effluents containing carbon-14 and is thus considered to be BAT;

- Tritium is minimised at source by a number of measures including the optimisation of boron and lithium concentrations, avoiding the use of boron for burnable poisons and control rods and reducing tritium production from secondary neutron sources. The majority of tritium is discharged into the environment in liquid discharges. Discharge in liquid form has lower radiological impact per unit discharged than discharge in gaseous form and is thus considered to be BAT;

- No practicable abatement techniques for the removal of tritium from liquid and gaseous discharges have been identified. The proposed operation of the EPR is consistent with international best practice for PWRs with respect to the management of effluents containing tritium and is thus considered to be BAT;

- Corrosion products, including isotopes of cobalt, are minimised at source through the implementation of a number of measures including limiting the cobalt content of materials and use of stellite, pre-operation oxidation of the primary circuit, zinc injection and optimisation of primary coolant chemistry. The majority of isotopes of cobalt (and other corrosion products) are removed from liquid and gaseous discharges by abatement measures such as filtration and thus converted into solid wastes, thereby minimising the radiological impacts of their disposal. This is consistent with the principle of ‘concentrate and contain’. The abatement measures used in the design of the EPR for corrosion products are considered to be BAT; and

- Noble gases and radioiodine production is associated with defective fuel whose occurrence is rare due to the high quality of fuel used and optimisation of reactor operation. Abatement of noble gases and iodine discharges is achieved by use of delay beds, filtration and demineralisation.

Decades of operating experience feedback demonstrate the efficiency of this development towards reducing discharges. An assessment of the treatment methods and long term discharges of various operating PWR has been carried out and shows that the waste water treatment systems that have been proposed for the UK EPR result in the lowest discharge values [UK EPR Generic Design Assessment – Assessment for the filtration and demineralisation systems of the CSTS, LWPS and SGBS systems [EDCME100828 Revision D. EDF. January 2011. (E)]

Beyond the GDA, EDF and AREVA will continue to keep the techniques used for the minimisation at source and the management and abatement of radioactive wastes under review. They will take account of operating experience feedback from EDF and other international operating experience feedback.

6. REFERENCES

UKEPR-0003-061 – PCER Sub-chapter 6.1 (E)

UKEPR-0003-062 – PCER Sub-chapter 6.2 (E)

UKEPR-0003-063 – PCER Sub-chapter 6.3 (E)

UKEPR-0003-064 – PCER Sub-chapter 6.4 (E)

UKEPR-0003-080 – PCER Sub-chapter 8.2 (E)

UKEPR-0003-080 – PCER Sub-chapter 8.3 (E)

UKEPR-0003-080 – PCER Sub-chapter 8.4 (E)

UKEPR-0003-110 – PCER Sub-chapter 11.1 (E)

NEEM-F DC 140 Revision A. Reduction in Primary Circuit Radioactivity SFAIRP Based Upon the Primary Circuit Materials. AREVA. March 2011. (E) .

NEEM-F DC 143 Revision B, Reduction in Primary Circuit Radioactivity SFAIRP Based Upon the Primary Circuit Chemistry AREVA. March 2011. (E)

EDCME100828 Revision D. UK EPR Generic Design Assessment – Assessment for the filtration and demineralisation systems of the CSTS, LWPS and SGBS systems. EDF January 2011. (E)

ENTERP090062 A EPR Primary Source Term of the EPR reactor, EDF, March 2009. (E)

EDEECH060283 – Study on transfer of carbon 14 from primary circuit to liquid effluents. Summary of PAL215 tests.(E)

ENTECH080059 – EPR – State of the art for abatement of carbon-14 in gaseous effluents, EDF.(E)

ECEF050113 – EPR Flamanville – Design study and validation of the main characteristics of the liquid waste processing system – TEU, EDF, March 2006. (E)

ECEF072084 – EPR Flamanville 3 - Summary report: specific design provisions related to EPR chemistry, February 2008. (E)

ENRE06109C – Radiation protection guidance for the design of pressurised nuclear equipment for French PWR reactors (E)

DEAPC080379 Revision A1, “Assistance and Recommendations for Tritium Management in NPP Systems. (E)

UK Strategy for Radioactive Discharges, 2006-2030, July 2009. (E)

European Commission, Guidance on the calculation, presentation, and use of collective doses for routine discharges, Radiation Protection 144, 2007. (E)

Generic Design Assessment of New Nuclear Power Plant Designs. Statement of findings following preliminary assessment of the submission by AREVA NP SAS and Electricité de France SA for their UK EPR design, Environment Agency, March 2008. (E)

NESH-G 2008 en 0123 Revision. A - Solid Radioactive Waste Strategy Report, AREVA, November 2008. (E)

Vance, J.N., Cline, J.E., Robertson, D.E., 1995 – Characterization of Carbon-14 generated by the Nuclear Power industry, TR-105715. EPRI, Palo Alto, California, Electric Power Research Institute. (E)

EPRI, Radioactive Liquid Processing Guidelines, 1011782, Final Report, November 2005. (E)

EPRI, NP 7386s V3P1-2, Radwaste Desk Reference, Volume 3: Part 1: Processing Liquid Waste, Final Report May 1994. (E)

EPRI, Radiation Field Control Manual, 1003390, Final Report, December 2004.(E)

EPRI, Primary Water Chemistry Guidelines, 1002884, 2005. (E)

EPRI, 2005 Interim Review of the Pressurised Water Reactor Chemistry Guidelines – Revision 5, 1009933, Final Report, December 2005. (E)

Zinc Injection Implementation at UK-EPR - ECEF110138 Revision A. EDF. March 2011. (E)

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

IAEA Technical Report 421 – Management of Waste Containing Tritium and Carbon-14, July 2004. (E)

Radiochemistry of Nuclear Power Plants with Light Water Reactors, K-H. Neeb, de Gruyter, 1997. (E)

Life cycle and management of carbon-14 from nuclear power generation, M-S Yim and F. Caron, Progress in Nuclear Energy 48 (2006) 2 – 36. (E)

Effluent release options from nuclear installations – Technical Background and regulatory aspects, OECD 2003. (E)

Technical Basis for Averaging C-14 Filters, Interim Report: Carbon-14 Source Term Analysis For Encapsulated Filter Waste Forms, EPRI, Palo Alto, CA: 2000. 1000849.(E)

Journal of Nuclear Science Technology 37 (9). (E)

Journal of Nuclear Science Technology 38 (109). (E)

“The techniques of the engineer”, G1815 Carbon Dioxide, P. Le Clourec, 2008. (E)

SZB/THR/042 - Review of the Control and Impact of the Discharge and Disposal of Radioactive Waste at Sizewell B Power Station (Information provided by British Energy Generation Limited for the review by the Environment Agency of authorisations under RSA 93. (E)

GDA UK EPR – Integrated Waste Strategy Document, UKEPR-0010-001. (E)

Rapport environnemental, social et sociétal 2007, Etablissement de La Hague”, AREVA NC.