Radiation Protection
CRPPH/R(2012)3
September 2012
www.oecd-nea.org

OECD

Good Practice in Effluent Management for Nuclear Power Plant New Build

A Report from the CRPPH Expert Group on BAT

AEN NEA

NUCLEAR ENERGY AGENCY

Unclassified **NEA/CRPPH/R(2012)3**

Organisation de Coopération et de Développement Économiques
Organisation for Economic Co-operation and Development **28-Sep-2012**

English - Or. English

Nuclear Energy Agency

NEA/CRPPH/R(2012)3
Unclassified

Committee on Radiation Protection and Public Health

Good Practice in Effluent Management for Nuclear Power Plant New Build

A report from the CRPPH Expert Group on BAT (EGBAT)

This document will be available as an electronic publication (available on www.oecd-nea.org/rp)

Edward.Lazo@oecd.org
NEA/RP

JT03327148

English - Or. English

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 34 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Republic of Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

This work is published on the responsibility of the OECD Secretary-General.
The opinions expressed and arguments employed herein do not necessarily reflect the official views of the Organisation or of the governments of its member countries.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 30 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, the Republic of Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

Acknowledgement

The CRPPH and the NEA Secretariat would like to thank Dr. Richard Doty for the high quality of this report.

TABLE OF CONTENTS

FOREWORD

At the 64th CRPPH meeting, in 2006, members of the Committee noted that in the current situation, the prospect of construction of new nuclear power plants looks more likely (or is already in hand) in member countries than for the last 15 years or so, yet during this time few member countries had seen new build. Therefore it was suggested that the CRPPH explore topics around optimisation of new build and, specifically, around Best Available Techniques (BAT) for effluent management and/or discharge abatement to assist member countries in preparing for the possibility of new build of a nuclear power plant. Based on this scoping work the CRPPH at its 65th meeting in 2007 established the Expert Group on Best Available Techniques to develop a report on this important subject.

The Group examined a wide range of issues relevant to discharge abatement but concluded that there was a lack of detail for discharge data for precursor plants ('generation 3') as may be applicable to BAT evaluations for new plants ('generation 3+'), and that improving this situation was a pre-requisite to understanding BAT for new plants. This proposal was accepted by the CRPPH although, partly due to the recognised difficulty of the task, it was agreed to proceed in phases, with CRPPH approval for each phase:

1. Comparison of discharge data for potential new build reactors.
2. Discussion of discharge data with stakeholders, e.g., reactor vendors.
3. Preparation of guidance on BAT, potentially with input into the Multinational Design Evaluation Programme (MDEP).

In order to pursue this work, the Group investigated effluent discharges and discharge management specifically from the precursors for the following reactor types, seen as the most likely for new build in some regions of the world: AP1000, CANDU, EPR, and ESBWR. (This characterisation should not be taken to imply that construction of new reactors units using, for example, Chinese, Indian, Korean, or Russian designs is unlikely in some regions of the world.) The Group discussed discharges with utilities operating precursor plants, including Konvoi and N4 plants (relevant for EPR reactors), the latest generation Westinghouse plants (relevant for the AP1000), ABWR plants (relevant for ESBWR plants), and the latest generation CANDU plants.

This report represents the first two steps in this process. It should be noted that some practical difficulties were encountered with this group following the 66th CRPPH meeting in 2008, such that work on this area was put on hold during much of 2008. However, the 67th CRPPH meeting reiterated the value of this work in May 2009, and the work plan was submitted to the CRPPH at its 68th meeting in May 2010 for preliminary review and approval.

The purpose of the report is to provide information on effluent management to plant designers and operators and also to regulatory authorities which will be making decisions on construction and operation of new reactor units. The report will consider the principles of optimisation related to projected doses to members of the public and the principles of "best available techniques" for effluent management during the life cycle of a new reactor unit. The results in this report should be considered limited and preliminary because there are both regional differences in approaches to plant design,

operation and regulation, and also because there will be opportunities for additional input to the topic on an ongoing basis. That input will arise from continued assessment of the operation of existing reactor units, the development of new techniques for more effective effluent management, emergent findings related to radiation science and the solicitation of input from relevant stakeholders.

The CRPPH notes that the Expert Group on Occupational Exposure (EGOE), which it established, published in 2010 a report titled "Occupational Radiological Protection Principles and Criteria for Designing New Nuclear Power Plants". While the focus of that report was on occupational radiation protection, there are elements in that report that address factors to be considered in the design of effluent treatment systems. That report is therefore a potential source of interest to those interested in discharge management for new reactor units.

1. INTRODUCTION

At the current time, it appears that there may be a wave of new construction of nuclear power plants. Indeed, new construction is already progressing in some member countries of the Nuclear Energy Agency (NEA) and elsewhere. However, there has been little new build of nuclear power plants in NEA countries for around fifteen years or more, while globally there seems to have been some consolidation of nuclear power plant designers and vendors. These factors of common timing and, increasingly, common designs, mean that many NEA member and other countries could be faced with similar challenges at similar times, whether because they are considering construction of plants themselves, or because they are in the vicinity of countries. A key topic for new build for national regulatory authorities is understanding what level of discharge to the environment may be expected from a new plant. Yet, despite the increased commonality of designs (themselves being evolutions of currently operating plants rather than being radically new), there seems to be little clear information on what may be expected from a new plant in this regard.

For the reasons given above, the Committee on Radiation Protection and Public Health (CRPPH) agreed at its 64th meeting that a study examining issues and information around Best Available Techniques (BAT) for management of discharges from new nuclear power plants should be carried out. This document is the first report resulting from this study.

2. OPTIMISATION IN RELATION TO EFFLUENT MANAGEMENT

The principle of optimisation of exposure of members of the public is described in the regulations of numerous countries and the policies of numerous facility operators. Those regulations and policies are based primarily on the principles as stated by two international organisations, the International Commission on Radiological Protection and the International Atomic Energy Agency.

2.1. International Commission on Radiological Protection

The system of radiation protection recommended by the International Commission on Radiation Protection (ICRP) is based on three general principles:

- Practices that are adopted are to be those for which a sufficient benefit to exposed individuals or to society offsets the radiation detriment that they cause. This is often called justification of the practice.

- Related to a source of radiation exposure, the magnitude of doses to individuals, the numbers of people exposed and the likelihood of incurring an exposure should all be kept as low as reasonably achievable, with economic and social factors being taken into account. Within that dose (or risk)-management process, there may be constraints placed on dose to individuals or risks to individuals to limit dose inequities which might otherwise occur as the various economic and social judgements are made. This is often called optimisation of protection related to a source of radiation.

- The total exposure of an individual resulting from exposure to one or more sources is to be subject to dose limits, to ensure that doses to individuals remain acceptable in normal circumstances. Similarly, there may be constraints established to control risks to individuals when there are potential exposures which may occur. This is often called the establishment of individual dose (or risk) limits.

Note that in a situation such as solid waste disposal, the occurrence of events may be probabilistic and the actual accrual of dose by an individual is not at all certain. It is for such cases that risk rather than dose limits may be used.

For purposes of this section of the report, it is the second principle mentioned above, that of optimisation, that is to be discussed in more detail. The ICRP states (for example, in its Publication 101, Ann. ICRP, 2006) that the optimisation process is to be an ongoing, forward-looking, iterative process involving (a) evaluation of the situation to identify the need for action, (b) the identification of potential protective actions to maintain exposures as low as reasonably achievable, (c) the selection of the option(s) to be implemented for the existent circumstances, (d) effective implementation of the selected option(s), and (e) review of the exposure situation to determine if additional or revised protective actions may be appropriate to maintain exposures as low as reasonably achievable. Recognizing that both quantitative and qualitative judgements are likely to be involved in decision-making regarding optimisation, the ICRP observes that a key element in the optimisation process is

maintenance of a questioning attitude regarding whether additional or revised protective actions can reasonably be implemented or whether no further dose reduction is reasonable.

In a broader context, the ICRP also notes that the optimisation process should be designed to consider the equity of individual dose distributions resultant from the protective actions implemented, verify that an active safety culture supporting optimisation is in place within the operating management of the facility and the regulator of that facility, and reflect the desirability of stakeholder involvement in assessments of effectiveness of the optimisation process.

Implementation of a graded approach is recommended, to consider both the magnitude of projected exposures and the complexity of the operations at the facility. Documentation of the decision-making process is to occur, to ensure transparency of the process and facilitate the conduct of reviews of the prevailing circumstances at future times. The operating management of the facility is to make decisions regarding the design, structure, and implementation of the optimisation process for the facility. The regulatory authority is to promote the use of a robust optimisation process and may require such a process be in place for the facility, as part of the review of the facility's application for a license to operate the facility and its ongoing review of the implementation of license conditions related to facility operation. Successful optimisation focuses on the effective use of robust processes to evaluate situations rather than on specific numerical results (presuming that doses actually received remain below the applicable dose limits).

In the case of management of radioactive effluents to the environment, the principles of maintaining doses to members of the public at levels which are as low as reasonably achievable and the principles of using "best available techniques" to control releases of radioactive materials to the environment are both approaches to optimisation and are complementary to one another. Both approaches are compatible in their objectives of limiting doses (risks) to humans, foreseeable effects on non-human species, and releases of radioactive effluents to the environment. Application of the optimisation principle, to achieve exposures that are as low as reasonably achievable (ALARA), is focused more explicitly on ensuring doses to individuals from a source (e.g., a new nuclear power reactor) are appropriately controlled, while application of the BAT principle is focused more explicitly at ensuring that effluent releases from that source are appropriately controlled, with more complete descriptive language discussing limitation of any concentration of radioactive materials in the environment. From the perspective of regional uses of approaches, one difference is that the phrase "best available techniques" tends to be used more often in western Europe, whereas the term optimisation is used more globally, although even the term "optimisation" may be more often discussed using the terms "as low as reasonably achievable" (ALARA) or "as low as reasonably practicable" (ALARP) in some regions.

The ICRP is developing recommendations describing more explicitly, protection of non-human species and the environment overall (as stated in ICRP publications 101 {Ann. ICRP, 2006} and 103 {Ann. ICRP, 2007}). As that work progresses, the ICRP uses the presumption that if humans are adequately protected from radiation and radioactive materials introduced into the environment, then non-human species and the environment are adequately protected from harm as well, and is working to develop a framework in which it can be demonstrated whether or not this assumption is valid in the situation being considered. The focus of ICRP recommendations is on limiting dose (or risk) resultant from radiation or radioactive materials introduced into the environment. Reduction of effluent releases arises out of the aim to reduce dose (risk) to levels which are as low as reasonably achievable.

2.2. International Atomic Energy Agency

The International Atomic Energy Agency (IAEA) establishes or adopts standards of safety for protection of health which a Member State may apply by means of its regulatory provisions related to nuclear and radiation safety. In its document on Safety Fundamentals (No. SF-1), the IAEA states a series of safety principles applicable throughout the lifetime of facilities. Several of these principles are listed below, as directly relevant to optimised effluent management at nuclear power facilities in relation to protecting people and the environment from harmful effects of ionizing radiation.

- The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risk.
- An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained.
- Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks.
- Protection must be optimized to provide the highest level of safety that can reasonably be achieved.
- People and the environment, present and future, must be protected against radiation risks.
- Protective actions to reduce existing or unregulated radiation risks must be justified and optimized.

These and the other principles described in SF-1 are to be considered by the Member States and practicably should be considered by other nations in establishing a nuclear power effluent management programme. The principles harmonise well with the recommendations of the ICRP related to optimisation for purposes of radiation safety.

In its Safety Guide No. WS-G-2.3 (2000), the IAEA amplifies its then-current statements on the regulatory control of radioactive discharges to the environment, limiting the scope of the safety guide to protection of human health. At the same time, the IAEA indicates that the acceptable protection of the environment is to be provided. The term “environment” is noted to include the protection of living organisms other than humans and also the protection of natural resources, together with consideration of non-radiological impacts on the environment. Describing the process of optimisation of protection, the IAEA in its Safety Guide states that evaluations should be made of the cost-effectiveness of available protective options and also that consideration should be given to the reasonably available means to minimise the generation of radioactive wastes or to eliminate them completely. Additional statements are used in the Guide to point out that trade-offs may be involved, for example in three areas: calculated doses to the public as compared to doses to workers involved in facility operations, calculated doses to current populations as compared to doses to future generations, and choices between options whose projected costs and projected impacts may be known with varying degrees of certainty.

In a document published in 2002 (IAEA-TECHDOC-1270), the IAEA describes the ethical considerations relevant to protecting the environment. The document was noted to be “one step in the development of a framework for the protection of the environment”, which framework is still in development to the current date within the IAEA, the ICRP, and other organisations. In the TECHDOC, five features are identified as being common among the ethicists studying the issue to that

date. Those features are as follows: (1) sustainability, (2) maintenance of biodiversity, (3) conservation, (4) environmental justice, and (5) respect for human dignity. It would be speculative to attempt to define the framework for environmental protection that may emerge from the ongoing efforts; the description above is meant only to describe that both human health and the environment are both being addressed in ongoing scientific evaluations.

3. BEST AVAILABLE TECHNIQUES (BAT)

3.1 Definition/Use for discharge management

"Best Available Techniques" is a term that is most commonly used in Europe and, as a term in its own right, largely stems from outside the nuclear industry. The term refers to adopting the best solution, within reason, to protecting the environmental as a whole, in major part by mitigating discharges. The focus of the BAT concept is on reducing effluent releases to the environment (where they cannot practicably be eliminated entirely). Reduction of calculated doses to members of the public may be expected to arise out of the reduction in effluent releases, but such reduced dose is not per se the focus of the BAT concept. Since discharges of radioactivity lead (or are calculated to lead) to radiation doses to members of the public, the application of the "ALARA" concept (reducing calculated doses to levels which are as low as reasonably achievable, likely by means of reduction in levels of effluent discharge) is complementary to application of the BAT concept (reducing discharges to levels which are as low as achievable given consideration to the cost of actions taken, and thereby likely reducing calculated doses). Even where the expression "BAT" is not used, terms with similar meaning may be found to apply. Therefore, although this report uses the term "Best Available Techniques", the principle this covers is widely used in describing evaluations of reasonably available options to reduce discharges during implementation of the optimisation process.

A requirement for Best Available *Techniques* should be distinguished from one for Best Available *Technology*, as the term "techniques" may be interpreted to include not only the equipment (technology) used, but also the processes and procedures associated with use of the equipment. In this report, "BAT" refers to Best Available Techniques. (This statement is not intended to imply that countries that may use the term "Best Available Technology" are focused narrowly on technology and are not considering effectiveness of implementation of the technology and associated discharge management practices.)

The key question addressed in selecting BAT is one of evaluating the trade-off between what can be done to further reduce discharges of radioactivity (remembering any wider implications of reductions, such as generation of other types of waste) and what a reasonable (or unreasonable) cost is for the operator, and by extension, society, to pay for that reduction. The term "reasonable" is a term requiring an inherent value judgement to be made; that is, for the decision-maker, there are social and ethical concerns which may factor into the decision on what a reasonable (or unreasonable) cost may be in a particular country or location. Therefore, a decision on what is BAT for a specific situation considers both the more traditional technical and financial factors regarding implementation of a technology or techniques (including treatment of the uncertainties regarding the probability of successful implementation of an emerging technology or techniques), but also considers social and ethical concerns that may be more difficult to put into a quantitative decision-making process. One example of where judgement may enter into the decision-making process as to what is "BAT" for a specific situation relates to the apparent, general preference for a "concentrate and contain" compared to a "dilute and disperse" approach to radioactive waste. This preference for limiting and monitoring releases of potentially harmful substances directly to the environment means that demonstrating the use of BAT (or an equivalent term) will be a key part of proposing, building and operating a new

nuclear power plant in those regions of the world where use of the BAT concept is expected. How discharge abatement requirements are applied will, however, depend on national approaches to regulation, for example, varying from a goal setting approach (with the operator planning to show that BAT are applied) to a more prescriptive approach (with key aspects of BAT decided upon and specified in regulations or regulatory authorisations). Close alignment of legal requirements is not believed necessary to be a prerequisite for useful harmonisation on BAT (or an equivalent term).

Determinations for the activity of what the BAT are, involve evaluations of the different approaches to controlling emissions to air, surface waters, and soil and/or ground waters. An integrated approach is to be used, to ensure that the environment as a whole is protected, and discharges are not merely shifted from one environmental medium to another. Such an approach would also ensure appropriate consideration of energy efficiency, waste management and accident prevention in an integrated manner.

Specific examples of the definition of Best Available Techniques follow.

Examples of International Definitions

Since BAT is open to many interpretations (e.g., what is “best”?), the term may be defined in more detail. Here, international policy and legislation may provide some guidance regarding interpretation of the term “BAT”. Although neither of the two uses of the term “BAT” as described below fully control radioactive discharges from a nuclear power reactor to the air, water, and ground, their definitions of “BAT” may be illustrative of the thinking of some countries regarding the term. Specifically, the descriptions are from the OSPAR Convention (related to control of radioactive materials released by facilities in the applicable signatory countries to the marine environment of the North East Atlantic) and the European Union Directive on Integrated Pollution Prevention and Control (which does not cover radioactive discharges from nuclear power plants). The definitions used in each of these are given below. The definitions underline that a broad view is to be taken in judging what is BAT.

OSPAR Appendix 1*

2. *The term "best available techniques" means the latest stage of development (state of the art) of processes, of facilities or of methods of operation which indicate the practical suitability of a particular measure for limiting discharges, emissions and waste. In determining whether a set of processes, facilities and methods of operation constitute the best available techniques in general or individual cases, special consideration shall be given to:*

   - *comparable processes, facilities or methods of operation which have recently been successfully tried out;*
   - *technological advances and changes in scientific knowledge and understanding;*
   - *the economic feasibility of such techniques;*
   - *time limits for installation in both new and existing plants;*
   - *the nature and volume of the discharges and emissions concerned.*

3. *It therefore follows that what is "best available techniques" for a particular process will change with time in the light of technological advances, economic and social factors, as well as changes in scientific knowledge and understanding.*

[4.]

5. *"Techniques" include both the technology used and the way in which the installation is designed, built, maintained, operated and dismantled*

**Convention for the Protection of the Marine Environment of the North-East Atlantic, text as amended through 18 May 2006, Appendix 1.*

IPPC**

"Best available techniques" means the most effective and advanced stage in the development of activities and their methods of operation which indicate the practical suitability of particular techniques for providing in principle the basis for emission limit values designed to prevent and, where that is not practicable, generally to reduce emissions and the impact on the environment as a whole:

- *"techniques" shall include both the technology used and the way in which the installation is designed, built, maintained, operated and decommissioned,*
- *"available" techniques means those developed on a scale which allows implementation in the relevant industrial sector, under economically and technically viable conditions, taking into consideration the costs and advantages, whether or not the techniques are used or produced inside the Member State in question, as long as they are reasonably accessible to the operator,*
- *"best" means most effective in achieving a high general level of protection of the environment as a whole.*"

***Directive 2008/1/EC of the European Parliament and of the Council of 15 January 2008 concerning integrated pollution prevention and control, Article 2.*

In November 2010, a directive on industrial emissions was finalised which will supersede the above-described IPPC directive. Per Directive 2010/75/EC of the European Parliament and of the Council of 24 November 2010, Article 3, the following definitions will apply as Member States implement the Directive over the next few years.

IED

"'Best available techniques', means the most effective and advanced stage in the development of activities and their methods of operation which indicates the practical suitability of particular techniques for providing the basis for emission limit values and other permit conditions designed to prevent and, where that is not practicable, to reduce emissions and the impact on the environment as a whole:

- *'techniques' includes both the technology used and the way in which the installation is designed, built, maintained, operated and decommissioned:*
- *'available techniques' means those developed on a scale which allows implementation in the relevant industrial sector, under economically and technically viable conditions, taking into consideration the costs and advantages, whether or not the techniques are used or produced inside the Member State in question, as long as they are reasonably accessible to the operator;*
- *'best' means most effective in achieving a high general level of protection of the environment as a whole.*

Examples of National Uses

While the above-mentioned international policy and legislation may be of substantial importance, especially in the cases where it directly relates to control of radioactive discharges to the environment, of more direct importance are the definitions of "BAT" that have been incorporated into national legislation and regulations, thereby directly impacting the licensing or permitting of a nuclear power reactor. Two national structures are described below, to illustrate definitions in use in the national context.

England, Wales, Scotland, and Northern Ireland

The basis for control of radioactive discharges in England and Wales is found in Environmental Permitting regulations adopted in 2010. In Scotland and Northern Ireland, control is via the Radioactive Substances Act of 1993. Specific to nuclear power plants, licence conditions relating to radioactive waste generation and discharge of wastes in controlled via the Nuclear Installations Act of 1965.

There is a regulatory requirement to demonstrate that doses are maintained as low as reasonably achievable. In England and Wales, optimisation is required of the facility operator through the use of licence conditions requiring the use of Best Available Techniques. In Scotland, optimisation is required of the facility operator through the use of authorisation conditions requiring the use of Best Practicable Means (BPM). The requirements to use BPM are considered by the regulator to be essentially equivalent to requirements to use BAT, as both have the objective of balancing costs with environmental benefits using a structured process.

The definition of BAT is very similar to that used in the OSPAR Convention. Specifically:

> Best Available Techniques (BAT) means the latest stage of development (state of the art) of processes, of facilities or of methods of operation which indicate the practical suitability of a particular measure for limiting discharges, emissions and waste. In determining whether a set of processes, facilities and methods of operation constitute the best available techniques in general or individual cases, special consideration shall be given to:

1. comparable processes, facilities or methods which have recently been successfully tried out;
2. technological advances and changes in scientific knowledge and understanding;
3. the economic feasibility of such techniques;
4. time limits for installation in both new and existing plants;
5. the nature and volume of the disposals concerned.

It therefore follows that what is "best available techniques" for a particular process will change with time in the light of technological advances, economic and social factors, as well as changes in scientific knowledge and understanding.

The above definition is taken from Regulatory Guidance Series, No RSR 2, "The regulation of radioactive substances activities on nuclear licensed sites", Environmental Permitting Regulations (England and Wales) 2010.

In context of the definition, the words in themselves are taken as follows:

"Best" means most effective in achieving protection from exposure to radiation as assessed against the range of detriments and benefits of further reductions;

"Available" implies that consideration has been given to whether the techniques being evaluated have been both developed on a scale which allows a facility operator to confidently implement the technique in the industry (e.g., at a nuclear power plant) and also determined to be economically and technically viable, considering the relative balance of costs and benefits of implementation.

"Techniques" means both the technology to be used and the way in which the facility is designed, constructed, operated, maintained, and decommissioned.

Consideration is to be given to the overall view of environmental, social, and economic factors. Non-radioactive detriments that may be introduced by implementation (or lack of implementation) of an option, need to be evaluated to ensure that overall detriment is not increased by the choice of technique chosen for control of radioactive material.

In England and Wales, there is a value judgement applied in terms of the balance of cost and benefit regarding implementation of a control measure. Practicably, options are screened out only when the costs to implement the option are considered "grossly disproportionate" to the benefit. Judgement is used on a case by case basis to determine what may be "grossly disproportionate". Precedent to date seems to suggest that cost to benefit ratios in the range of 2 to 10 may meet the threshold as "grossly disproportionate" for cases involving lower and higher risks to members of the public, respectively, but with multiple factors to be considered in the case by case decisions. With that understanding by the Environment Agency, implementation of an otherwise technically viable option is to be undertaken unless the costs are grossly disproportionate to the benefits gained.

In making decisions regarding BAT, the following principles are to be addressed: sustainability of development, waste hierarchy and waste form, the precautionary principle, and the proximity principle. Also, the degree of assessment (and documentation requirements) are to be proportional to the magnitude of the issue being evaluated and should consider good practices observed at other facilities which appear to be applicable to the facility under consideration. The precautionary principle implies that decision-makers should take into account a social responsibility to protect the public if there is a reasonably credible though uncertain risk that might result in serious or irreversible damage. The proximity principle implies that waste should be managed or disposed as close as reasonably possible to its point of origin, given the available options.

The above commentary is taken (paraphrased and abridged) from "Best Available Techniques (BAT) for the Management of the Generation and Disposal of Radioactive Wastes, A Nuclear Industry Code of Practice, by the Nuclear Industry Safety Directors Forum, as Issue 1, December 2010 and commentary found within RSR 2, "The regulation of radioactive substances activities on nuclear licensed sites", Environmental Permitting Regulations (England and Wales) 2010.

Sweden

Regulations in Sweden that limit releases of radioactive materials from nuclear installations are consistent with the Swedish environmental quality objective of maintaining a safe radiation environment. The limitation on releases is based on the principle of optimisation (ALARA) and the use of best available techniques. In this context, BAT is defined as follows:

> A best available technique is the most effective measure available to limit the release of radioactive substances and the harmful effects of the releases on human health and the environment which does not entail unreasonable costs.

For the purpose of regulating releases from nuclear power plants using the ALARA and BAT concepts, limitation is accomplished through the restriction of dose to pre-defined critical groups. A licensee is to demonstrate that calculated doses from discharges are below 0.1 mSv/a to the most exposed individual (member of the critical group). Reference and target values are then set. The reference value is based on release rates of radionuclides that signify optimal performance of the reactor and its systems affecting generation, elimination, or delay of discharge of radionuclides into

the environment. The reference value determined by the facility operator is reviewed by the regulatory authority. The target value determined by the facility operator reflects the operating management's plan in terms of further reduction in discharges to the environment, taking into account the continuing use of the BAT concept. The defined target value is associated with a timeframe within which the operating management believes it can achieve the target value. The licensee is to annually report to the regulatory authority regarding measures implemented or planned to achieve the target value. If the (higher) reference values are exceeded, the licensee is also to report on plans to achieve the reference values.

The definition and commentary above for Sweden is taken from SSI FS 2000:12, Regulations on the Protection of Human Health and the Environment from the releases of Radioactive Substances from Certain Nuclear Facilities, Swedish Radiation Protection Authority, 2000. In 2009, the regulatory reference changed to SSMFS 2008:23, Radiation Safety regulations on the protection of human health and the environment of discharges of radioactive substances from certain nuclear installations, Swedish Radiation Protection Authority, 2009, for which an official translation into English is not yet available.

"Costs" to be considered in selecting BAT include those related to ensuring that risks to society and/or the environment are reduced overall, not just transferred from one to a different segment of society. Similar to the commentary provided in the section above on the International Atomic Energy Agency, examples may be (a) ensuring that an inordinate increase in risk to workers does not occur due to the implementation of techniques selected to reduce environmental risk, (b) ensuring an appropriate weighting to calculated doses to the current populations as compared to calculated doses to presumed populations far into the future, and (c) ensuring an appropriate consideration and weighting to individuals near the facility likely to receive higher levels of calculated dose (however small) as compared to individuals farther from the facility and likely to receive lower levels of calculated dose. Also, the degree of certainty or uncertainty regarding ability to effectively implement a mitigation option is to be considered. In the applicable cases, transboundary implications of calculated doses need to be considered in addressing societal factors. While substantial transfer of risk may be unlikely in most situations, an integrated assessment of risks from facility operations is recommended.

Decision-aiding and decision-making for determining BAT applicable to the local circumstances of a nuclear fuel cycle facility are complex. The ideal methodology may utilise a systematic, evidence-driven, transparent means of demonstrating optimisation occurs, proportional to the scale of the issue to be resolved. Options are characterised, qualitative and quantitative assessments of those options are conducted, knowledge gaps and uncertainties are addressed or acknowledged, stakeholder opinions are sought, and a decision is taken. However, a wide range of environmental factors are to be considered, and radioactive emissions and radiation doses are not the sole contributors to risk to workers or the public. Cost-benefit analyses, multi-attribute analyses, or other means of balancing costs and benefits are difficult when costs of option implementation and benefits (reduced radiation doses) are translated to monetary equivalents and/or attributes to be considered can be described only qualitatively and with a level of subjectivity. Studies may be inconclusive and/or similar cost-benefit outcomes may be interpreted as basically the same, given the uncertainties involved. The decision-making thus may involve the application of judgement. In such cases, principal factors (e.g., constraints and assumptions) leading to the decision for the specific facility should be documented, especially for those cases where the decision results in implementing an approach that some may interpret as being disproportionate to the risks and potential impacts posed.

3.2 BAT in the context of optimisation in authorising discharges

Discharge limits for radioactive materials from nuclear power stations have been set traditionally at levels limiting calculated radiation dose to a member of the public to no more than a specified value (that is, avoiding exceeding an annual dose limit). In some countries, a dose constraint has also been defined by the regulatory agency for a member of the public, set at a value below the dose limit, determined by the agency to be an achievable annual calculated dose given available options to maintain equity in individual dose distribution and given the desirability of setting a margin for doses due to regional and global sources and from exempted sources.

[Note: Monitoring or measurement of dose directly to members of the public is often difficult if not impossible, in part because the magnitudes of effluent releases are often near the lower limits of detection of effluent monitoring equipment and because there exist natural background levels of radionuclides in the environment which may mask and exceed the levels calculated to be resultant from the effluent releases in the environment. That implies that environmental concentrations, land-use data, and dose coefficients related to calculated intakes of materials may be used in calculating doses to members of the public. This basis in use of environmental concentrations is one way that the "ALARA" and "BAT" approaches are complementary. More information is provided in later sections of this report.]

Optimisation processes (using the complementary "ALARA" and "BAT" approaches) are used by facility operators, in consultation with regulators and other stakeholders where appropriate, to determine if lower levels of discharges (and resultant calculated annual doses) are reasonably attainable. The outcome of those planning processes during design, construction and operation is that the actual discharges themselves have most often been substantially lower than those values specified as regulatory discharge limits based on ensuring that annual dose limits (generally on the order of 0.1 – 1 mSv/a) are not exceeded. This favourable outcome may reflect successful use of optimisation processes, reducing both discharges and doses calculated to be received by members of the public. The difference between retrospectively determined (actual) discharges and the prospectively determined, authorised discharges (which is termed by some to be an "operating overhead" or an allowance for operational flexibility) is to have been considered in the authorisation process, to give the operator flexibility to cope with non-routine events, unplanned maintenance and minor deviations from the design parameters. Allowance for that flexibility is prudent; however, several concerns regarding involvement of stakeholders may be foreseen, and plans should be developed for addressing the concerns effectively (e.g., by preparing appropriate educational materials or establishing appropriate processes to handle emergent situations in a timely and effective manner).

Three examples follow, describing situations for which prepared educational materials may be helpful. First, if the discharge rates are consistently low enough or the "operating overhead" is consistently large enough, then some stakeholders may perceive that there conceivably could be reduced pressure for applying emerging "best available techniques" to give continual improvement in discharge rates. Second, there is the potential that such continued large differences between prospectively authorised discharge rates and retrospectively determined (actual) release rates may lead to some stakeholders perceiving that the prospectively defined values apparently give regulatory authorisation for an operator to consistently discharge greater quantities of radioactivity than they will in practice discharge, from the reported actual release rates. Thirdly, and on the other hand, too low a difference between actual and prospectively authorised rates, or too low an operating overhead, may result in operators breaching an authorised discharge rate when carrying out reasonable and necessary activities that may have negligible radiological impacts, but may be expected to be discussed by stakeholders as a regulatory violation, with all attendant implications of such a violation. Internal to

the operating management and the regulatory staff, there may then be expected to be associated investigations of inadequate performance and the delineation of appropriate corrective actions.

If it is accepted (using one approach to development of authorised discharge limits) that actual discharges should nominally be a substantial percentage of the prospectively authorised discharge rates (that is, the operating overhead should be reasonably small), then the challenge is to devise a transparent and consistent approach to setting authorised discharge rates at levels that are stringent enough to guarantee a high level of performance in relation to discharges, while giving operators the flexibility they need to conduct normal, acceptable operations and/or to respond to non-routine plant conditions without infringing their discharge authorisations.

To give an idea of what such an approach might look like, an example is given below; probably a realistic aim in this area would be general acknowledgement of the principles which some countries might wish to adopt in promulgating their own approaches to implementation. As an example of another approach, a system of averaging over the authorisation period could be used, analogous to that allowed for occupational doses (e.g., for workers to receive up to 50 mSv in a year provided that the average over 5 years does not exceed 20mSv per year).

Possible Methodology for Dealing with Differences between Actual and Authorised Discharge Rates

Note: a diversity of approaches or methodologies may be found to be appropriate in different countries and/or regions. A country may, for example, set authorised discharge rates based on ensuring that annual dose limits for members of the public are not exceeded and/or ensuring that annual dose constraints are not exceeded. In such case, the operating management of the facility would use the applicable "ALARA" and "BAT" (where required) processes to ensure effective implementation of technology and good management practices. Regulators would review the operator's use of its optimisation program to ensure that the program is implemented according to program procedures.

Principles for use in those countries where the regulatory authority establishes (or approves the operator's application for) authorised discharge limits at levels below those required by consideration of annual dose limits

- The authorised discharge rates should be based on the levels of discharge that the operator has determined to be optimised for plant operation, considering the discharge abatement techniques available to the operator. Note that there may be differing approaches to declaring a situation to be in regulatory violation, dependent on whether exceeding a specified authorised discharge rate results in a regulatory violation or whether an approach such as the reference value used in Sweden is used, whereby exceeding a reference value may result in investigation but not necessarily a regulatory violation.

- The authorised discharge rates should provide for necessary operator flexibility based on fluctuations or trends in the levels of discharge over time, that the operator has substantiated will occur in routine operation and anticipated operational occurrences (e.g., start-ups, leaks and malfunctions expected to occur during the operating lifetime of the plant), even though BAT have been applied.

- The difference allowed between actual discharges and prospectively authorised discharge rates should be kept near the minimum necessary for the normal operation of the plant (routine operations and anticipated operational occurrences).

- The authorised discharge rates are **not** set at levels corresponding to the boundary between acceptable and unacceptable radiological impact or to a calculated value of dose to members of the public equal to the annual dose established in traditionally set generic discharge limits. In particular, they do **not** correspond to the dose limits contained in national or international legislation. The application of BAT should in all likelihood have eliminated any proposals which would give rise to calculated doses approaching or exceeding such limits before the stage is reached when authorised discharge rates are established.

Implementation

- Plant Identification. Plant components which substantively contribute to discharges should be examined to identify which individual ones or groups warrant being subject to authorised discharge rates. Authorised discharge rates usually apply at the point(s) where the discharge leaves the facility. Establishing authorisations on individual plants, or groups of plants, where reasonable, rather than just on the overall site, increases the effectiveness of controls; provides transparency for operators, regulators and the public; and allows BAT reviews to be carried out in a staged manner. Note: the approach of setting individual plant discharge rates rather than site rates may differ from other practice. For example, in Europe, the Sulphur Content of Liquid Fuels Directive allows a "bubble" limit to be set for sulphur dioxide discharges from oil refineries. Consideration of the number of reactors at a site may be appropriate in evaluating the value of any dose constraint set by the operator or authority at a level below the applicable annual dose limit. As the number of reactor units built on a site increases, then it may be reasonable to evaluate if the likelihood of other radiation sources being established in the nearby surroundings practicably decreases. Such an evaluation may result in defining a site-based dose constraint that uses less margin to the applicable dose limit.

- Establish Base Case Discharges. For a new plant, base case discharges (BCD) are derived from the operator's flow sheet data in the developed design of the reactor, excluding accident conditions: usually twice (x2) those estimated by assuming continued optimal performance of relevant reactor systems, to allow for operational flexibility. Typical base case discharge data for one or more similar plants elsewhere in the world (believed to be using good practices in its effluent control systems) should be provided by the operator and verified by the regulator.

- Choosing Radionuclides for Authorised Discharge Rates. The radionuclides or groups of radionuclides for which authorised discharge rates will be set should be chosen on the basis of whether they meet one or more of the following thresholds. Specific to defining groups of radionuclides, one or more of the radionuclides within a group should feasibly be reliably measurable; a group shares relevant characteristics in production, effluent sampling or dosimetry; and/or the group contributes to calculated dose. (In the context of defining an authorised discharge rate, the criterion selected is likely to be a release or dose rate pre-selected for a radionuclide or set of radionuclides; for example, a criterion for radioiodines may be a calculated annual individual dose not to exceed 0.01 mSv.) If no threshold is exceeded, then there may be no need to establish an authorised discharge rate for the radionuclide or set of radionuclides.

a. The calculated dose to a member of the critical group from the BCD is greater than the agreed criterion. National preference may establish whether the dose is to be calculated based on generic, conservative assumptions (sometimes called conservative "screening") or site-specific, realistic calculations;

If additional thresholds are deemed appropriate for use, the following thresholds may be considered:

b. The collective dose from the BCD is greater than the agreed criterion;

c. The BCD are greater than the agreed criteria;

d. The regulator considers that the discharges of a reliably measurable radionuclide, or group of radionuclides, is a good indicator of plant performance, process control or any other contribution to the demonstration of the use of BAT

- Setting the Authorised Discharge Rates. The authorised discharge rates should be set equal to the BCDs for the radionuclides, or groups of radionuclides, identified, unless a different methodology is deemed reasonable for the specific facility.

- Authorised discharge rates are usually set without prescribing the use of a specific technology or management practice. The technical characteristics of the facility, its geographical location, and the local environmental conditions may all be considered in defining the level of discharge that is authorised.

- Transboundary implications of authorisations are to be considered, with affected countries consulted as appropriate should substantial transboundary effects be projected.

- Authorisation permits should contain sufficient specificity as to monitoring and measurement methodological outcomes to ensure that release data are collected at adequate frequency, reported data are of reasonable accuracy and detail, and checking for compliance with the permitted values can reliably be accomplished.

If different countries deploy essentially the same types of nuclear power plants it will be useful to be able to demonstrate a common approach internationally, even if the differences between authorised and actual discharge rates themselves differ from country to country.

3.3 Effluent management in the context of new build

Overview approach

As has already been alluded to, the market for nuclear power reactors is more globalised than in the past, so that now more than ever, countries are likely to be hosting the same design of plant. This section aims to investigate how the concepts of dose optimisation for members of the public (ALARA) and of best available techniques (BAT) are being applied and, in particular, how common approaches, positions or consensus may be achieved, and how areas for further discussion may be identified.

Optimisation processes undertaken in relation to (or at least considering similar factors as addressed in) the recommendations of the International Commission on Radiological Protection are used extensively in virtually all countries in the world, and the requirement for adoption of an

optimisation programme is a part of the regulatory system in virtually all countries of the world. Commonly, for effluent management systems being designed for use in new nuclear power plants, an optimisation program to be used by the designer and operator would include the following elements:

- A structured, on-going, forward-looking process that evaluates reasonably available technological and process/procedure options for management of discharges of radioactive materials to the environment;

- A means of identifying those discharge-management options to maintain exposures of members of the public at levels which are as low as reasonably achievable, such as via examination of systems and practices available at operating plants similar to the facility being planned and via discussions with major technology vendors for the nuclear power industry;

- A documented selection process for options to be implemented, including consideration of possible complicating factors such as potentially increasing occupational exposures while reducing doses to members of the public, and proportional to the complexity of the radiation safety issues being evaluated;

- A robust engineering programme for implementation of the selected options in the design of the facility;

- A robust pre-operational planning programme to ensure effective implementation (operations, maintenance) of the selected options; and

- A means of continuing reviews of the overall design and construction process to ensure that anticipated exposures to members of the public from the facility are at levels which are as low as reasonably achievable. A part of that continuing program is maintenance of an active safety culture throughout the design and construction phases for the new facility.

The designer and/or operator of a proposed facility will calculate estimated doses to members of the public surrounding the proposed facility, using applicable generic and available site-specific data on land use, population densities, and other factors relevant to the calculation process. The designer and operator need to ensure that those estimated annual doses comply with the applicable dose limits. Additionally, in many countries, there are also either dose constraints, dose targets, or other applicable values which are to be met by those estimated doses forthcoming from the planning process. Whether that additional value must be achieved, is to be achieved absent a variance granted by the regulatory authority, or is recommended to be achieved, may vary from country to country, depending on the regulations applicable to each country. In any case, there is the expectation of a review of the design and anticipated effluent releases by the applicable regulatory authority, to ensure that an optimisation programme has been used in the design process.

Before describing examples available in the international governmental and regulatory arenas more directly related to the application of the BAT concept, a brief description (amplifying that information provided directly above) will be given regarding the role of the facility operator in maintaining effective effluent management programs at a facility. The amplification is intended to transition from the commentary above on optimisation using an ALARA approach to plant design toward a commentary more directed to design using a BAT approach, but still reflecting commonalities in process across regions of the world applicable to an ALARA-based approach to design of effluent management systems.

The Role of the Operator

To address the potential for impact on members of the public and the environment, aspects of plant operations related to discharge control and abatement need to be considered effectively and efficiently in plant design, construction and operation. In this report, both the selection of technologies to be used and the establishment of management practices to be employed are described.

- Initial selection of technologies and discussion of anticipated practices (to be used by the operator of the facility in implementing selected technologies) is anticipated to be performed at the time of plant design and pre-operational planning. As the design process progresses, some changes to technologies and anticipated practices may occur as the plant design is optimised relative to the purposes for which the facility is planned. The design process may be influenced by characteristics of the site on which the facility is to be constructed. For example, if there are already nuclear fuel cycle facilities on the site, such that the new facility is to be an additional facility at the site, the design process is to address how if at all the discharge abatement technologies (existing or planned) on the site may be affected by the addition of the new facility.

- The operator should communicate to the appropriate regulator the results of the design and pre-operational evaluations, to inform the regulator's assessment of discharge-management measures that are in place or planned to be put in place.

- Selection of technologies and establishment of practices are not "once and done". Periodic or repetitive evaluations of available technologies and good practices are expected. Comparison of technologies and practices used at the subject facility with those used at well-operated, similar plants globally is envisioned, to determine if reduction in discharge rates is reasonably feasible (i.e., emissions/wastes may be reduced significantly without imposing excessive costs). At a time that the operator proposes an extension to the operating lifetime or the power-generating capability of the facility is one opportune to time for a re-evaluation of techniques to use at the site for discharge abatement.

- The operator should communicate to the appropriate regulator, the results of those re-evaluations over time, to assist the regulator in determining that appropriate revised discharge-abatement measures are in place or planned to be put in place.

- The operator is to develop plans to employ suitable discharge monitoring, environmental monitoring and assessment methods, both to inform their own decision-making and to ensure compliance with discharge authorisations/permits and dose limits. Results of environmental monitoring may also be used to verify the effectiveness of the effluent management procedures and processes used at the site. The operator is also to develop plans for measures to ensure equipment reliability is as intended and that fuel integrity is maintained. Both of those program elements tie directly to effective discharge management of the proposed facility.

International Harmonisation

In this sub-section of the report, harmonisation of approaches to effluent management is discussed. Because a significant amount of harmonisation exists regarding the use of the optimisation process as recommended by the International Commission on Radiological Protection, the commentary focuses more on harmonisation regarding use of the BAT concept as a complementary approach to effective consideration of discharge abatement.

The European Union Integrated Pollution Prevention and Control (IPPC) Directive

Essentially, the IPPC Directive concerns minimising pollution from various industrial sources throughout the European Union. Operators of industrial installations are required to obtain an authorisation (environmental permit) from the authorities in the relevant EU country. Tens of thousands of installations are covered by this Directive. Releases of radioactive materials to the environment from nuclear fuel cycle facilities are not under the purview of the Directive.

An IPPC permit is a single integrated permit which deals both with emissions to all environmental media and also to the environmental management of the facility. The IPPC permits aim to prevent or reduce emissions to air, water and land, reduce waste and use energy/resources efficiently. These permits must contain conditions-based BAT as defined in the Directive, to achieve a high level of protection of the environment as a whole.

The Directive requires the European Commission to organise an exchange of information between Member States and the industries concerned with BAT, associated monitoring and developments in them. This is performed via the European IPPC Bureau. The Bureau organises this exchange and produces BAT reference documents (BREFs) which Member States are required to take into account when determining BAT generally or in specific cases.

A BREF contains a number of elements leading to conclusions as to what are considered to be BAT in a general sense for the industry sector concerned. The definition of BAT requires that the technique is developed on a scale that allows implementation in the sector. The evidence to support a technique as a best available technology can come from one or more plants applying the technique successfully somewhere in the world. In some cases, even pilot projects can provide a sufficient basis, with adequate evaluation. The standards which are to be achieved, in controlling emissions, are included in the BREFs.

The BREFs aim to inform the relevant decision makers about what may be technically and economically available to industry in order to improve their environmental performance.

OSPAR

The OSPAR Convention was established in 1992 and came into force in 1998. It combined and up-dated the 1972 Oslo Convention on dumping waste at sea and the 1974 Paris Convention on land-based sources of marine pollution. The OSPAR Convention does include releases of radioactive materials to the marine environment of the North East Atlantic within its scope.

Contracting Parties are required to take all possible steps to prevent and eliminate pollution and to take the necessary measures to protect the maritime area against the adverse effects of human activities. To this end, the convention requires, inter alia, the application of: the precautionary principle; the polluter-pays principle; and the use of best available techniques (BAT) and best environmental practice (BEP), including, where appropriate, clean technology.

The requirement for the use of BAT stems from OSPAR being a regional partner in the implementation of the Global Programme of Action (GPA) for the Protection of the Marine Environment from Land-based Activities. The GPA requires the establishment of criteria for assessing and/or reporting on the use of BAT to reduce discharges of radioactive substances.

Every four years, those Contracting Parties with a nuclear industry submit a national report on progress made on the implementation of BAT in nuclear facilities to minimize and,

as appropriate, eliminate radioactive discharges to the marine environment. These reports adhere to the OSPAR "Guidelines for the submission of information about, and assessment of, the application of BAT in nuclear facilities". The reports (should) include the following for each such facility:

i. A description of the plant, technology and procedures used to minimize and, as appropriate, eliminate liquid, gaseous and solid discharges into the Convention area;

ii. Improvements in such waste treatment plant, technology and procedures in the previous four years;

iii. Time series of discharges for at least the previous four years;

iv. Reasons for changes in discharge patterns;

v. As appropriate, proposals for future reduction in discharge levels.

OSPAR does not produce BREF-type documents. The detail of reporting differs between Contracting Parties, types of facilities (nuclear reactors, fuel reprocessing, fuel fabrication, and research reactors) and also between similar types of facilities.

The process provides OSPAR Contracting Parties with a forum to allow discussion/debate on BAT in nuclear facilities in other countries. However, in keeping with the OSPAR objective for "progressive and substantial reductions of discharges", the application of BAT in this work concentrates on the reduction/elimination of liquid discharges to the marine environment.

US Nuclear Regulatory Commission (Multinational Design Evaluation Program)

The Nuclear Regulatory Commission (NRC) in the U.S. takes part in the Multinational Design Evaluation Program (MDEP), a 10-nation initiative that is cooperating on specific safety design reviews and exploring harmonization of regulatory requirements and practices. The national regulatory authorities from Canada, China, Finland, France, Japan, the Republic of Korea, the Russian Federation, South Africa, the United Kingdom, and the United States are working together as part of MDEP. The International Atomic Energy Agency (IAEA) participates in the generic issues of MDEP.

Currently, the MDEP has five working groups, three of which are concentrating on generic issues in the areas of Mechanical Codes and Standards, Digital Instrumentation and Controls, and Vendor Inspection Cooperation. Two design specific working groups are cooperating on the reviews of AREVA's EPR and Westinghouse's AP1000 designs that are being licensed in several countries.

The MDEP participants share information regarding issues important to MDEP members (i.e., as raised and discussed in MDEP Working Group meetings) and plan to share information with other regulators when it is appropriate to do so. The primary means of MDEP to communicate with other non-MDEP regulators is through interactions with the Committee on Nuclear Regulatory Activities' (CNRA) Working Group for the Regulation of New Reactors (WGRNR). In addition, the MDEP plans to make information available to other stakeholders, as appropriate, in the form of MDEP Common Positions and other products that document MDEP participant's evaluations and recommendations on best regulatory practices.

Expansion of MDEP membership and further cooperation with other regulators is a possibility and an issue that will be discussed at the upcoming MDEP Policy Group meeting at which the heads of each of the 10 national regulatory authorities are invited to attend.

While BAT (or equivalent) has not been explicitly mentioned in the overall goals of MDEP, the MDEP may include some component addressing the use of BAT. In fact radiation protection issues surrounding the EPR designs have been discussed as part of the EPRWG activities. It is important to note that the objectives of MDEP includes making each participant's regulatory authority stronger in its safety decisions and supporting the sovereignty of each regulator in its authority and responsibility to take timely decisions on safety of new reactors.

Discussion

An examination of the different international contexts indicates that there is an emerging common understanding of BAT, at least within the European community if not worldwide. The NEA, with its report published in 2003 titled "Effluent Release Options from Nuclear Installations", has noted that the IPPC system of BAT reference documents provides a level playing field of recognised environmental performance standards across the EC, although radioactive emissions from nuclear facilities do not fall under the scope of the IPPC (nor emerging Industrial Emissions) directive. Such a common understanding would assist regulators (and operators) in learning and potentially improving plant performance with respect to discharge abatement, particularly since it appears that a new wave of nuclear reactors would be composed of a limited number of (internationally deployed) designs.

However, any considerations for international harmonization regarding the use of BAT should give due care to the legislative differences in the nuclear countries in the world, since national approaches and priorities may differ.

Any further work to set basic obligations and general principles on BAT should build on work that has already been done, for example, in this document and through the IPPC Directive, OSPAR, and national legislation already adopted. The variation in reporting under OSPAR (which only concerns discharges affecting the marine environment) should be reduced with the use of common designs and also indicates the strength of a "BREF" type approach. Work in this area would be underpinned by an understanding of discharges, comparability of which is discussed in this document.

There are issues which may productively be addressed, to further understand the complementary nature of the ALARA and BAT approaches and to further clarify the potential advantages of adoption of the BAT concept outside the European community. Examples of such issues to be considered may be as follows:

1. For a "BREF" type of approach to be fully useful, the regulators, designers, and operators of nuclear fuel cycle facilities need to have confidence in the approach taken to defining which technologies may be confidently applied to an industry sector in a technically and economically viable manner. The current approach is Eurocentric, and additional clarity is desired regarding effective means to apply the methodology across regions of the world.

2. There are different results which may be forthcoming in evaluations using factors defined to include value judgements regarding terms such as "reasonable", "unreasonable", "excessive", and "grossly disproportionate". The connotations of the terminology appear to be somewhat inherent from the language used in the international policy documents underlying the uses of approaches such as "ALARA" and "BAT". Discussion regarding the value judgements and their bases and interpretation may be appropriate in the various

national (and perhaps international) settings as changes to regulations are considered. Harmonisation may be possible, or the lack of full harmonisation from country to country or region to region may at least be better understood from such discussions among the various stakeholders in each country and, where appropriate, region.

3. There remain ongoing scientific evaluations as to how non-human species and the environment overall are to be protected, and if the protection of humans at dose limits adopted by regulatory bodies, in part from the most recent recommendations of organisations such as the International Commission on Radiological Protection, is sufficient for the protection of non-human species and the environment overall. Some regulatory authorities and industry organisations may wish to discuss if adoption of an approach such as the BAT concept is timely or potentially may be premature, as results of those evaluations become available over the next several years. This discussion should consider how application of the precautionary principle may inform the decision-making, as there remains a lack of full clarity of information on the potential detriment to the environment, however small, at current effluent release levels from nuclear power reactors.

4. Regulations in many countries require an optimisation (ALARA) programme regarding dose to members of the public to be established and maintained by operators of nuclear fuel cycle facilities. The regulator then reviews the implementation of the programme as it is defined by facility processes and procedures. This approach ensures that an effective optimisation process is implemented and an optimum result is achieved, but it does not mandate that discharge rates to the environment be reduced on an ongoing basis absent finding that reduction can be accomplished in a cost-effective manner. Under the BAT concept, there appears to be more of a mandate to plan to reduce discharge rates, even when cost-effective means to do so have not been identified but may possibly be identified. There is a relationship between this issue and the first and second items identified in this list; the intent would be to establish clarity regarding the level of effort required of the operator to attempt to identify potentially feasible means of reducing release rates. Another way to state one component of the issue is a request for clarification regarding whether an optimisation process is effective only when the result is an ongoing measurable reduction in discharge rates to the environment. A related component of the issue is how the balance between nuclear safety, industrial safety, radiation safety, and environmental safety is measured and judged; for example, in the potential trade-off between increased occupational dose and reduced discharge rates. Similarly, a discussion may be appropriate regarding the balance between power generation from a facility and discharge rates from the facility; that is, on the balance between what might be reduced power generation to meet needs of the consumers of that power and reduced discharge rates to the environment with attendant power consumed by the effluent treatment systems.

Existing practice as guidance; avoiding "mistakes" in defining BAT for new plants

In designing and constructing a new nuclear power plant, a level of reliance will be placed on using techniques considered to be BAT at plants in operation. The discharge rates for the plants in operation will be assessed by the designer and operator of the proposed new plant. The anticipated discharge rates for the proposed new plant, resultant from the proposed technologies and practices for the new plant, will be compared to those discharge rates of the existing plants. If the comparison is favourable (same or lower discharge rates for a plant of equal or greater rated thermal power), one tentative conclusion that might be reached is that BAT are defined for implementation in the new plant.

There are two issues related to such a conclusion. The first is that the assumption is made that the existing plant(s) to which comparison is being made has indeed implemented BAT. If that assumption is invalid, the tentative conclusion is in error. The designer and operator of the proposed plant should discuss with the plant(s) in operation, the results of the latest assessment of whether the plant in operation is using BAT and whether changes in technologies or management practices are planned to reduce discharge rates. Having that additional information can inform the designer and operator and allow for better decision-making.

The second issue is that emerging technologies and practices need to be assessed, to determine if the anticipated discharge rates for the new plant can be reduced via implementation of those emerging technologies and practices, without excessive costs (including costs which might be incurred if a regulatory certified design is being proposed to be changed). Even if the plant(s) in operation could not reasonably incorporate an emerging technology or practice, it may be possible to reasonably incorporate the emerging technology or practice in a plant still in the design or early construction phase.

Two other considerations should be noted. The first is that proposed new plants are expected to have operating lifetimes of 60 to 80 years. It is reasonable to presume that substantial equipment and technology changes/replacements may be made over that time. Incorporation of a very specific technology may appear to be appealing and optimum for near-term emissions control. However, incorporation of a more flexible equipment space and technology may result in lower cost and lower discharge rates over a longer period of time. An example may be some of the radioactive waste solidification systems built into some plants, which had to be redesigned at high cost to incorporate more flexible and more manageable systems using technologies emerging as the plants were constructed.

The other consideration to mention is that to make modifications to plant systems usually takes substantial time to design, may take substantial to time to achieve regulatory approval for the change, may take substantial time to install in the plant, and may raise plant safety questions to resolve as the transition from “old” to “new” occurs. It is not uncommon for modifications to take on the order of 3 to 5 years or more to implement in an existing plant or a plant already into the late design or construction phase. Recognition of that fact should be considered as assessments are being done to determine if BAT are in place (or being put in place), if relatively new or emerging technologies may become the BAT, or if an assessor suggests a somewhat experimental approach (e.g., “try this; if it doesn’t quite work out, we’ll make an additional change”).

Radiation and other potentially harmful aspects of operation

Radioactivity is one hazardous property amongst others (e.g., toxicity). Other substances, e.g., chemicals, may have hazardous properties (but are not radioactive) and are frequently used in nuclear installations. These compounds are often used in a process to address particular safety or radiological protection issues. For instance the use of chemical conditioning of the primary system may generate releases of boron, which is used as a neutron absorbent to limit the reactivity of the reactor core. Application of BAT should not ignore this kind of situation. There are often technical answers to these types of questions (e.g., increase of the recycling of the aerated primary liquid effluents, use of boron highly enriched in ^{10}B). The chemical conditioning of the secondary circuit is another example of including chemical releases (in this case, an example may be morpholine) in consideration of BAT.

The concept of optimisation (application of “ALARA” for radiation doses) is very well developed and usually well executed in the field of radioactivity. If executed properly, the risks from other agents other than radioactivity are considered in the dose optimisation process. There is therefore

a clear tie with taking into account chemicals associated with the radioactive releases as a consideration for the definition of BAT. It is not sufficient of course to consider just chemical releases in defining BAT; as with the ALARA concept, numerous factors may apply (e.g., industrial safety for plant workers). Notably however, without any association with radioactive releases, nuclear installations can nevertheless generate chemical releases that are similar to industrial ones.

These chemical releases can be due to the intentional use of some chemicals. Some cooling systems are using chloride compounds to prevent the development of microbiological organisms. This kind of treatment is probably not different in other industrial sectors and so practice in this area from other industries could be used, for example, through the corresponding Best Reference document prepared by the EU through its IPPC Directive. Another example of intentional addition is the introduction of depleted zinc (usually in the form of an oxide) as a means of reducing or controlling occupational exposures by controlling the build-up of a radionuclide such as cobalt-60 on in-plant piping. Introduction of the zinc may serve its intended purpose but also may affect the concentrations of cobalt in waste media. An additional example is the introduction in boiling water reactors of series of agents, such as iron, hydrogen, and oxygen, to maintain more optimum conditions for protection of reactor core internals and associated recirculation piping, but with potential effects regarding industrial safety in maintaining certain plant components and radionuclide concentrations in waste media. There are evaluations ongoing in plants on a day to day basis to determine if water treatment system changes or changes to the concentrations of materials in the reactor coolant may be appropriate to effect positive changes in nuclear, industrial, radiation, or environmental safety. A recent example may be the use of specialty clean-up resins with the aim of reducing occupational exposures, but with an effect on concentrations of radionuclides in liquid effluents and solid waste media. While these evaluations may be made by operating plants staffs and implemented on operating plants, there are certainly implications on the design of new reactor units, as design engineers and plant operators analyse the integrated effects of such changes on all aspects of plant safety, including effects on discharge rates.

In other cases the chemical releases from the nuclear installations are not generated by the intentional introduction of some substances. They may be due to the design of the installation itself. For instance the type of condensers used at a nuclear power plant can influence (over a large range) the level of copper released.

Furthermore, chemical or microbiological releases may as well be generated by an “amplification” of the concentration of existing contaminants. This situation is often met in open cooling systems where evaporation leads to an increase of the concentration of some contaminants. For microbiological organisms, cooling systems may act as a kind of reactor leading to the development of undesirable agents (e.g., legionella).

These considerations, and others, lead to the conclusion that BAT cannot be defined for radioactive releases without considering the impact that may be induced on other kinds of releases (or intakes) or risks.

The example of the reduction of copper levels from condensers demonstrates that it is important to have a global approach. To reduce the level of copper releases, stainless steel condensers came into use in nuclear power plants. These components have been identified as a favourable location for proliferation of NF amoebae. To limit their quantities in water to an acceptable threshold, an operator was obliged to treat their systems initially with bleach, and then with monochloramine. Specific licenses had to be issued to deal with releases linked to these treatments.

This example illustrates that any choice may have adverse effects. Integrating the different technical points of view is the only way to avoid negative consequences or to limit or balance them to the degree technically feasible.

Questions on the identification of non-optimised approaches to discharge management

The identification of technologies and practices that together are not optimised for a facility is important. Such identification enables both the operator and the regulator to focus on selection of technologies and practices that together would become optimised for the facility, when implemented. To assist the operator and regulator in evaluating techniques being used and determining whether they are optimised for the facility, a checklist approach may be used. The following questions are examples of those used in such a checklist. The questions are written in a form designed to be more directly amenable to determining if techniques are BAT but their use, potentially with slightly modified wording, may also be considered in implementing the ALARA concept. Note should be made that having a technique available to the operator implies that its use would not compromise the non-environmental aspects of safety (i.e., nuclear, industrial, or occupational radiological) of the facility or the purpose for which the facility was constructed (e.g., for a nuclear power plant, the safe and reliable generation of electricity).

- Are there techniques that have been identified as being available to the operator (demonstrated to be technically and economically successful at similar facilities) that would substantially reduce either the facility's total release rate of radioactive materials or the release rate for a group of radionuclides?

- Are there techniques under development that may be expected to become available to the operator in the reasonably near term <u>and</u> for which the operator may contribute to the successful completion of that development without the expenditure of excessive costs? An affirmative answer to this question does not mean that the techniques in place are not BAT, but may imply that the operator should consider making such a contribution to the development of the technique and should consider implementation of the technique when available.

- Are the technologies and practices in place consistent with the objectives of

  i) optimising the generation and disposal volumes and/or activities of wastes

  ii) relying on concentrating and containing wastes rather than diluting or dispersing them

  iii) progressively reducing discharges of radioactive materials when reasonably practicable?

- Consistent with the above questions, and bearing in mind the likely costs and benefits of a measure, are there techniques available to the operator which better address in total the following considerations:

  i) the use of low-waste technology

  ii) the use of substances that are lower in their potential to be hazardous

  iii) the furthering of recovery and recycling of substances generated and used in the facility and of waste, where appropriate

iv) comparable processes, facilities or methods of operation that have been tried with success on an industrial scale

v) technological advances and changes in scientific knowledge and understanding

vi) the nature, effects and volume of the discharges from the facility

vii) the commissioning dates for new or existing installations

viii) the length of time needed to introduce the best available technique

ix) the consumption and nature of raw materials (including water) used in the process, and energy efficiency

x) the need to prevent or optimally reduce the overall impact of the discharges on the environment and the risks to it

xi) the need to prevent accidents and to mitigate to the extent practicable, the consequences of an accident on the environment

xii) the information published by national and international organisations on best available techniques, associated monitoring, and developments in them?

The considerations listed in the last question above are based primarily on wording in Annex IV of Directive 2008/1/EC of the European Parliament and of the Council of 15 January 2008 concerning integrated pollution prevention and control (which as stated previously is not directly germane to discharges of radioactive materials).

3.4. Optimisation for emergencies and incidents

This section seeks to explore whether the application of optimisation processes, including the concept of Best Available Techniques, in the area of emergency and incident planning management is adequately covered through safety management. At the very least, a high degree of overlap between optimisation or "BAT for emergencies" and safety management might be expected, since protection of the public and workers in emergencies and incidents is favoured by containment of radiation, applied through the defence in depth principle. As stated previously in this report, optimisation as described in documents of the ICRP includes the presence of an active safety culture within the operating management of the facility and the regulator of that facility. The presence of a robust safety culture per se may be more directly observed in the means used to prevent the occurrence of incidents and/or emergencies which may present a risk to workers and the public, while the presence of a robust process of controlling or mitigating risk (potential doses to workers and the public) may be more conventionally seen as the province of the optimisation process.

Physical analyses and management strategies for severe accidents and design basis accidents are covered by the NEA Committee on the Safety of Nuclear Installations' Working Group on the Analysis and Management of Accidents (GAMA). Attention is focused on event prevention, especially for design basis accidents (e.g., loss of coolant, LOCA) but mitigation is also considered (e.g., means of containing the core in the event of a severe accident). Attention will extend, for example, to the amount of flammable or explosive gases generated and suitable material choices (e.g., high quality concrete). The uncertainties and probabilistic studies of incidents are mostly carried out by the NEA Working Group of RISK (WGRISK). Work is also carried out through the NEA Working Group on

Structural Integrity (WGIAGE) on structural integrity through compilation of information on various risks (like the occurrence of seismic events), construction materials and, for example, pipe failures. In an optimisation or BAT context, it may be said that this shows event "prevention" to the maximum reasonably achievable degree as being the "best available technique".

In the event of an incident, including relatively minor ones, the operators have their emergency operating procedures. Similarly, the managers have their decision support tools and guidelines for emergencies. These are based on the analyses of the known phenomena, uncertainties and the plant features. In real emergencies, the action taken will often depend on the manager's choice based on their knowledge, experience and interpretation of the individual situation.

In this respect, incident reporting schemes, like the IRS and projects run by the NEA, capture knowledge that may be written into operational procedures or decision support tools like emergency management guidelines. However, because of the judgment aspect, one should be somewhat cautious with "Best Available Techniques" (in the sense of prescribing operating procedures) in the area of emergencies. Although analysis and planning is the primary key to success in incident and emergency management, one also needs to prepare for decision making in the face of uncertain and surprising situations (in recent literature, this is sometimes called "resilience engineering"). Therefore, in fact, BAT would consist of using incident reports together with other analyses (like probabilistic risk analyses, thermal hydraulics, structural integrity, knowledge on fuel behaviour, etc.) to generate and implement "integrated lessons learned" for safety management. These management aspects should be covered by radiological protection and nuclear safety in general, while the NEA runs several incident databases that give information on incidents/system failures that have occurred.

Nuclear safety management operates through prevention of incidents through defence in depth, redundancy and diversity with suitable guidelines or decision support tools should an incident occur, operating under the broad principle that minor releases of radioactivity are permissible to prevent a larger release (e.g., controlled venting to avoid a catastrophic loss of containment). In general, therefore, control of risk (doses) or optimisation and "BAT" for incidents and emergencies is implicit in application of effective nuclear safety management.

In a European context, the Western European Nuclear Regulators' Association has carried out a benchmarking exercise on implementation of defined safety features of nuclear power stations, with a broad aim of achieving harmonisation at a basic level on reactor safety. [Harmonization of Reactor Safety in WENRA Countries, Western European Nuclear Regulators' Association Reactor Harmonization Working Group, January 2006] This work reinforces the preceding discussion in that avoidance of loss of containment of radioactivity is included in the safety benchmarking, which also emphasises the importance of management systems, knowledge management and staff competence.

3.5. Management systems, including "quality assurance" (QA)

Adoption of an ALARA or BAT approach is not solely a case of having technology; even assuming the technological approach adopted is optimised or "best", it still needs to be applied effectively. Hence, inherent within the optimisation process and within the definitions of BAT are considerations of both the technology used and the way that an installation is designed, built, operated, maintained and dismantled. As well as reinforcing the "cradle-to-grave" aspects discussed in this report, this consideration of process as well as equipment indicates that management systems have an important role in effectively implementing the ALARA and BAT concepts.

Good management is essential for achieving high levels of safety, environmental and commercial performance and for promoting a strong culture in these areas. These needs require comprehensive

management systems to be in place to deliver appropriate leadership and decision-making, and a capable organisation to be in place which has an ability to learn lessons from internal and industry operating experience. Such systems should include arrangements for:

- Establishment of strategies, policies, plans and standards;
- Control of resources and contractor support, including control of organisational change and succession planning;
- Self-assessment, quality assurance, continuous improvement, and the training and qualification of personnel;
- Record-keeping and regulatory compliance.

Management systems are a strong component of nuclear power plant safety, thus there is considerable overlap with the field of nuclear safety. Assuming that strategies, policies, plans and standards needed to ensure the effective use of optimised techniques are adopted and maintained, the organisational approach to their management should largely be ensured by nuclear safety, since management failure in an area not directly related to safety may be taken as an indicator of ineffective management generally, thus implying weaknesses in safety management.

Nevertheless, while regulatory surveillance through safety concerns may generally cover management and organisational culture, attention may need to be paid to relevant, specialised organisational sub-units, including contractors, which may be outside the 'mainstream' management processes and organisational culture. In any case, it will be important to ensure that relevant strategies, policies, plans and standards are in place and applied. For example, maintenance of key equipment and relevant documentation should be verified, as programmes such as equipment reliability are inherently tied to programmes for discharge management.

In some countries, the commitment to high-quality management practices is demonstrated by an organisation’s obtaining and maintaining accreditation to International Standards Organisation (ISO) standards such as the ISO 9000 family of standards on quality management and/or the ISO 14000 family of standards on environmental management. Achievement of high quality (whether demonstrated by ISO accreditation or other means) may be expected of the organisations performing release measurements for the operator or independent, verification measurements for the regulator.

4. EXAMPLES OF DISCHARGE MANAGEMENT

4.1 Dose Limits, Dose Constraints, and Authorised Discharge Rates

Differences in approaches between countries

The International Atomic Energy Agency has reported on the experience of eighteen (18) Member States in establishing radioactive discharge authorisations. That experience is reported in the Appendix to "Setting Authorized Limits for Radioactive Discharges: Practical Issues to Consider", IAEA-TECDOC-1638, IAEA, 2010.

Level of conservatism

The authorised discharge rates of the Loviisa and Olkiluoto power plants are activity-based rates but they are based on the dose limit set in Finland for a member of the public (0.1 mSv per year, site-specific limit). All the significant exposure pathways have been taken into account when calculating the authorised discharge rates. It is not possible to be exact about the level of conservatism in calculated doses to members of the public without detailed comparison with other calculations. However, the aim is to have conservative assessments of calculated dose, but those calculated doses are not to be too conservative (that is, "realistically conservative" calculations are the objective).

From experience at Sizewell B in the United Kingdom (UK), while there is a dose constraint of a maximum individual dose of 0.3 mSv per site, or 0.5 mSv total from multiple adjacent sites, authorised discharge rates are all activity-based. The activity-based authorised discharge rates are set based upon past performance, that is, what has been achieved, without consideration of the actual dose received by a member of the public (except to the extent that the dose constraints were met). Presumably, if the aforementioned constraints were to be challenged, then this approach would change.

In some few cases, the authorised discharges for a year are based upon a rolling twelve-month total of reported monthly discharges. This basis in a rolling calendar period does not align exactly with the calculation of annual doses, which use calendar years, but has the objective of preventing large discharges in successive months at the beginning and end of calendar years.

Authorised discharge rates are reviewed and revised if necessary approximately every five years. The operator is encouraged to propose revised authorised discharge rates based upon past performance and any changes in operations or abatement techniques planned for the coming five year period. For sites with multiple sources of discharge, the addition of a new source or the decommissioning of a source may result in a proposal for revised authorised discharge rates. Other considerations may for example be the potential for changes in discharge rate as the plant ages, or operating conditions change, or site thermal power is proposed to increase (or potentially decrease), or upgraded equipment is installed.

The achievable discharge rates having been determined, the next step is to determine the reference person (critical group) by use of a land use or habit survey of the plant environs. The dose to the reference person is calculated for planning purposes as if discharges take place at the authorised discharge rates. Usually, different reference individuals are identified for different discharge routes –

for disposals to air, sea (water) and by direct exposure. Use of this approach results in a conservative assessment of calculated dose to any member of the public from the conservatively estimated discharges from the plant This is both because discharges are assumed to be at the limiting values and also because any summation of doses across different reference individuals is conservative.

The starting position adopted by the Environment Agency of the UK, when setting authorised discharge rates during the most recent revision of those rates, was that the maximum authorised value for each nuclide group should not exceed the annual peak discharge during the previous ten years, plus 25%. Exceptions were allowed where operators were able to demonstrate that this was overly restrictive.

The following table is taken from the Decision Document, published by the Environment Agency to explain to the public the terms and conditions of the discharge authorisations granted to British Energy sites. The table shows the calculated dose to the reference individuals with discharges occurring at the respective authorised discharge rates.

Table 5.11.10 Predicted doses by radionuclide to groups most exposed to discharges and disposals from Sizewell B Power Station at Environment Agency new limits

Doses (microsieverts/year) (1)			
Radionuclide	**From aerial discharge (infant)**	**From combustion (infant)**	**From liquid discharge (adults)**
Tritium as HTO	0.28	0.005	0.009
Carbon-14	3.6	Not Applicable	Not Applicable
Beta particulate as Cobalt-60	0.006		
Noble gases (Kr-85)	0.002		
Halogens I-131	0.77		
Caesium-137	Not Applicable		0.1
Other		0.002	1.1
Total	**4.6**	**0.002**	**1.2**

Notes:-

1 Groups most exposed to gaseous and combustion are the same. The group most exposed to liquid discharges is distinct and, therefore, the doses from gases/combustion and liquid in this table are not added together.

The examples described above demonstrate different methods of setting authorised discharge rates. In the Finnish example, the authorised rates are based on annual dose limits, whereas in the UK example, the authorised rates are based not only on dose constraints but also on assessed achievable performance. This shows the sorts of variability that is seen in methods used in different countries to defining authorised discharge rates for a reactor (or reactor site or set of nearby reactors).

In some countries, the authorised discharge rates are activity-based, while in others they are nominally dose-based, but translated to activity-based values with the use of calculational methods performed by the operator or regulator. In virtually all countries, there is a level of conservatism built into the process because any summation of doses calculated to be received by different reference individuals for given exposure pathways in inherently conservative. Depending on the method used to define the presumed actual discharge rates as compared to the authorised values or as compared to the limits of detection on the effluent monitoring equipment, an additional level of conservatism occurs.

4.2 Measurement techniques/Minimum Detection Level (MDL)

Discharge measurements are (directly or indirectly) based on laboratory measurements of samples (liquid, gaseous and particulate filters) taken from the discharge pathways. Examples of

detection limits can be seen in the Finnish discharge report to the European Commission (EC) shown in Annex 3.

For some radionuclides, discharge rates are such that reasonably accurate and precise measurements of the rates can occur, because the rates are higher than those MDLs of the effluent monitoring instrumentation. For other radionuclides, discharge rates are such that there is more uncertainty regarding the actual discharge rates, because the rates only occasionally are higher than the MDLs of the effluent monitoring instrumentation. In other cases, actual discharge rates are quite uncertain and are estimated based on the MDLs for the radionuclide or based on calculated ratios of the radionuclide of interest to another radionuclide whose discharge rate can more easily be measured.

4.3 What is included in measurements?

In Annex 3, an example of the radionuclides measured via effluent monitoring systems is provided.

Measurements for carbon-14 in effluent discharges are being performed for more units world-wide than had been the case a few years ago. Where measurements are being made, dose assessment is facilitated when the chemical form(s) of the release(s) is known. Reactor type and other factors may affect the relative amounts of carbon-14 being discharged via pathways to the air as compared to pathways to water and pathways involving solid wastes. Where measurements are not performed for carbon-14 in effluents, the use of scaling factors may be applied to estimate release rates, which tend to be conservatively estimated using that methodology.

4.4 What radionuclide groups are used?

In Finland, authorised discharge rates for radionuclide groups are set for noble gases (Kr-87 equiv. Bq) and iodine (I-131 equiv. Bq) to the air and other gamma emitting radionuclides (Bq, other than H-3) to the sea. It is reasonably common world-wide to have results reported for noble gases, for halogens (including the iodines), for particulates excluding tritium and often for tritium itself and for carbon-14. It is not necessarily true that there are authorised discharge rates for each of those categories, but the operators and regulators may jointly agree on the groups of radionuclides for which effluent-monitoring results are expected to be reported. In many cases, the discharge rates for radionuclides of short half-life (e.g., a few days or less, such as eight days or less) are not reported, because for virtually all of such radionuclides, there would be essentially no dose to a member of the public (or biota) and thereby essentially no adverse impact on the environment.

Example Information from Sizewell B

To provide substantially more specificity regarding a discharge monitoring program, additional information is provided as Annex 4 on the program for Sizewell B station in the UK. The intent is not to recommend that the program described need be followed in other countries as is. Rather, the intent is to indicate the sorts of questions and information that need to be considered in developing a discharge monitoring program of high quality. Plant procedures, practices and audit/assessment methods need to be developed to ensure that a high-quality effluent-monitoring program is established and maintained.

Where the site is required to measure a specific radionuclide, such as iodine-131, the requirement in the UK is to use best practicable means to sample and measure the quantity discharged. However, the monitoring requirements for certain radionuclide groups are less specific, using terms like “Noble Gas” or “Beta-emitting radionuclides associated with particulate matter”, where the actual

radionuclides are not specified and the value reported depends upon the technique used to sample and measure. For the less-specific radionuclide group requirements, the technique to be used is mutually agreed by the operator and regulator and is recorded in the discharge authorisation. Changes to the procedure are by agreement between the operator and the regulator. Such techniques are pointed out in the sections below. Where appropriate, sections of the discharge authorisation or the site's Techniques Documents are reproduced in Annex 4, to ensure that complete information is given.

Limits of Detection (LoD) are not specified by the regulator but are implicit in some of the techniques used. Where the requirement to use Best Practicable Means to sample and measure applies, there is also an implicit expectation to minimise the LoD. Where disposals are measured as less than the LoD, the LoD is used to assess the disposal. That is, there are no "zero" discharges reported unless the discharge flow rate is reduced to zero, which is possible for the Gaseous Radioactive Waste System (GRWS). Use of this methodology results in a conservative estimate of the activity actually released by the reactor site. See also section 6.2 of this report for some additional information on the interpretation of measurements which are below the LoD at some frequency.

There are alternatives to use of discharge rates assumed to be at the level of the LoD to calculate doses to members of the public (or otherwise assess potential impact on the environment). Those alternative measures may be used in other countries or regions of the world, with agreement between the facility operators and regulators. If statistical measures are available, for example, to analyse measured data to provide estimates of discharge rates (and estimates of uncertainty in those discharge rates) at levels between zero and the LoD, then those estimates of discharge rates may be useable in impact assessment, as mutually agreed between operators and regulators. If there is evidence to support other estimates of discharge rates based, for example, on radionuclide production rates and scaling factors, then such estimates may be useable in impact assessment, as mutually agreed between operators and regulators. One attribute of importance is documentation of the process used to estimate discharge rates for the radionuclides or sets of radionuclides being reported as released to the environment. There is additional discussion of this matter is section 6.2 of this report.

4.5 Reporting approaches

The reporting of discharge rates to the environment is often called out in regulations to be performed by the operator on an annual basis, but in some cases may be required to be performed on semi-annual, quarterly or even more frequent bases. In some cases, estimates of the uncertainties associated with those reported discharge rates are also to be presented. Where associated calculated doses to members of the public are to be reported, there is also a requirement to provide information so that the dose estimates may be replicated by the regulator or other affected stakeholder. For example, meteorological data for the site area, or flow rates in a river to which discharges are made, are to be provided.

In the same or a separate report, the operator may be required to report the results of environmental monitoring conducted around the facility. Those data provide additional information on radionuclide concentrations in the environment and may thus provide information on any build-up of radionuclides in the site area as a result of discharges occurring over a multi-year period. Notably, the environmental data may also provide a means of validating or verifying the results of the discharge monitoring, with analysis of the two sets of data. There is of course also the potential that the environmental data may be used as input to revising estimates of discharge rates reported in the future, as the environmental data may demonstrate that less or more radioactive material is found in the environment than would be expected, based on the reported discharge rates.

To add some specificity to the general comments above, additional information from Finland and the United Kingdom is described here. In Finland, power plants report the discharges quarterly and annually to the regulatory agency, STUK.

In the UK, discharges of radioactivity are reported every month to the Environment Agency. These "returns" are placed by the Agency on the public register and so are available to the general public.

The Environment Agency also maintains the "Pollution Inventory" into which operators enter more detailed isotopic breakdowns on an annual basis.

British Energy summarises its environmental performance in its annual Environmental Report. This will be subsumed into the EDF Sustainability Report in the future.

The practice in other countries is similar to those described for Finland and the UK. In some cases (e.g., the United States), the annual report is a compilation of discharge data based on analyses for calendar quarters. Reporting of discharge monitoring data on at least an annual basis is recommended.

4.6 Continuous and non-continuous discharges to the atmosphere; seasonal and other considerations

In Finland, aerosols are mainly discharged in connection with annual maintenance and refuelling outages. Discharges of noble gases and C-14 are more uniform across the calendar year.

At Sizewell B in the UK, ventilation air from the Radiologically Controlled Areas is disposed of continuously, via the Unit Vent Stack and the Radwaste Building HVAC Stack. (Ventilation air from the Reactor Building is normally isolated, except when people require entry.)

The Gaseous Radioactive Waste System Stack (GRWS) is designed to operate continuously, removing excess hydrogen and radioactive gases from the Reactor Coolant System (RCS). This enables hydrogen pressure control and assists in maintaining plant dose rates "as low as reasonably achievable" (ALARA), especially in the presence of "leaking" fuel. Operating the system continuously also prevents an accumulation of other radioactive species, such as carbon-14, in the RCS, from which they can then accumulate on ion exchange beds. The System uses a Carbon Bed Delay System to abate short-lived noble gas and iodine radionuclides, rather than delay tanks used at many other plants.

In terms of discharge profiles, Annexes 2 and 3 provide examples of data from Sizewell B and from Finland. Annex 5 provides additional information compiled in the United Kingdom and United States for operating units found in various regions of the world.

As noted above, many plants world-wide have at least some continuous releases to the atmosphere. There may also be tanks on a reactor site designed specifically to contain radioactive materials for decay before release to the atmosphere. Releases from such tanks are not continuous but are scheduled events during the year. Also as noted above for Finland, discharge rates for some reactors are affected by whether the reactor unit is in operation or is shut down for a maintenance or refuelling outage. Releases of particulate matter tend to increase after unit shut down for units undergoing substantial outage work activities.

As additional plants begin to measure C-14 in their discharges, there may be variability discovered over the period of unit operations. The data will be subject to review as the data are reported for units in additional countries and additional regions of the world.

4.7 Continuous and non-continuous discharges to aquatic systems; seasonal and other considerations

Radioactive waste water is discharged to the sea at the Loviisa and Olkiluoto power plants. Liquid discharges mainly happen in connection with annual outages.

At Sizewell B in the UK, liquid discharges are to the sea. The length of the discharge line (to a point 600 metres off-shore) coupled with local sea currents are such that there is no tide dependence on discharges.

Most of the Liquid Radioactive Waste Tanks at Sizewell B have quite small volumes, such that one or more tanks need to be discharged every day. The volume per month increases towards the end of the fuel cycle, following a reactor trip, and during refuelling outages.

For releases to aquatic systems (e.g., river, lake, or sea), it is more likely that a given reactor site releases specified volumes of material from tanks, such that the releases may occur once to several times a week rather than continuously. There are also plants which are “zero liquid discharge” plants, with no planned releases to the aquatic environment. This concept may sound very appealing in the context of reducing discharges to the environment; however, caution is advised in that there are numerous factors to consider in evaluating the practicality and the other potential impacts on workers and the environment due to adoption of a “zero liquid release” policy.

The description above does not address the potential for discharges to the ground water on site. Such releases tend to occur as an “event” due to malfunctioning equipment, such as a leak in a piping system used to transport a fluid containing radioactive materials from one point in the plant to another. There is also the possibility of a portion of a planned discharge to the atmosphere depositing on the ground surface of the site from dry or wet washout from the discharge to the atmosphere. For some countries, there is a separate component of the plant discharge monitoring programme to assess the potential for unplanned releases to the ground water on the reactor site. For some individuals and agencies, this may be a component of the effluent monitoring programme, while for others, a component of the environmental monitoring programme for the site. In any case, the objectives are to detect in a timely manner substantial discharges of radioactive material to ground water on site, and to enable assessment of the situation or development of supplemental monitoring if necessary to quantify the potential for release offsite and to assist in developing means to reduce the potential for release offsite.

5. EXAMPLES OF BAT APPLICATION

Several reports have been written to describe summaries of available discharge abatement techniques for liquid and gaseous wastes. Examples are the following:

- "Effluent Release Options from Nuclear Installations, Technical Background and Regulatory Aspects", OECD, 2003, ISBN-92-64-02146-9, and

- "Combined Methods for Liquid Radioactive Treatment, Final Report of a Co-ordinated Research Project 1997-2001", IAEA, 2003, IAEA-TECDOC-1336.

Various techniques shown to be proven in limited applications and found to be in continuing development are presented in such reports. The challenge to the plant operator and to the regulator is to assess the practicable viability of adoption of such techniques (and those otherwise identified) for full plant-scale waste treatment. Listed below are several examples of possible techniques for use for the limited purposes of discharge abatement for specific radionuclides or groups of radionuclides. Annex 6 contains a listing of some abatement techniques which have been or may be found useful in some specific applications.

5.1 H-3, C-14

Tritium (H-3) is a natural product which is produced by cosmic rays but which is also produced in nuclear power plants. While tritium has a very low dose coefficient and is unlikely at low concentrations to result in substantial harm to humans or to biota, there is a need to consider its discharge rates from nuclear sector facilities and to reduce those discharge rates to the extent reasonably achievable. The percentages of the releases of tritium to the atmosphere as compared to releases to aquatic systems may vary by the reactor type. There is no technique demonstrated to be both effective and reasonably cost effective for abatement of tritium in the discharges of most types of nuclear power plants. Continuing evaluation of the development of abatement techniques is warranted. A potential complicating factor is that techniques which may continue to contain tritium in plant systems may potentially increase occupational doses to workers at such a plant.

Because tritium can be produced in nuclear power plants by both the fissioning process and also by boron-related activities, the abatement of tritium source terms created via the boron-related activities is to be considered. An example of the latter is as follows:

Cook units 1 and 2 in the United States addressed the issue of boron recycling which was substantially contributing to the tritium discharge to the atmosphere prior to 2007. New technology and practices were implemented in 2007 to discharge the distillate from a Boric Acid Evaporator instead of recycling the tritium-containing distillate back to the primary coolant. The spent fuel pool tritium concentrations dropped by 60%, and the annual discharges of tritium to the atmosphere dropped by 50%. The use of depleted boron in the primary loop has resulted in reducing the amount of boric acid bottoms to be recycled. This change has also contributed to the reduction of tritium in gaseous effluents.

Applicable to carbon-14, the measurement of total beta in discharges addresses the release of carbon-14 to the environment. (At many facilities, carbon-14 discharges are measured directly.) As with discharges of tritium, the percentages of carbon-14 released to the atmosphere as compared to releases to aquatic systems may vary with reactor type. However, means to address the potential for abatement of carbon-14 discharges continue to warrant study.

5.2 Fission and Activation products

Releases of noble gases from the plant tend to occur more predominantly to the atmosphere rather than to aquatic systems, but neither pathway can be ignored in evaluations. In some plants, abatement of discharges to the atmosphere comes primarily from decay of radionuclides in piping systems designed for substantial decay to occur before discharge to the atmosphere occurs. Discharges of noble gases to the environment may be reported in a radionuclide-specific manner or may be reported on an activity-equivalent basis for a selected radionuclide such as Kr-87 or Xe-133.

Regarding releases of fission and activation products (that are not removed by resins or other means to become a component of the solid wastes from the site), discharges to both the atmosphere and aquatic systems may be anticipated. Monitoring of the separate discharge pathways would be appropriate.

Relating to both sections 5.1 and 5.2, opportunities should be sought for scientists and other affected stakeholders to discuss means that appear to show technical and economic promise for reducing discharges of the various radionuclides and/or sets of radionuclides to the environment. From such discussions may arise techniques that may reasonably be implemented to reduce discharges.

6. COMPARABILITY OF DISCHARGE DATA

6.1 In the context of national regulations

Discharge monitoring programmes and discharge reporting requirements differ from country to country. Operators of nuclear power plants and other nuclear fuel cycle facilities within a country must of course respond to the regulations and strongly consider the regulatory guidance applicable to the country in which they operate a specific facility. This situation results in data reporting which meets the current needs of the regulatory authority of each country. This situation also results in some difficulties in analysing discharge rates world-wide and enabling the optimisation of discharge rates at plants within the various countries based on an understanding of the performance of plants world-wide in terms of effluent management and control. As countries consider the desirability of harmonisation of monitoring and reporting criteria, there is expected to be a movement toward further harmonisation of approaches to monitoring and reporting. In section 6.2 there is discussion of some areas in which harmonisation may lead to performance improvement.

6.2 Issues that may affect comparability

The level of discharges from a nuclear power plant is critical for judging what is BAT or is optimal from the perspective of maintaining doses to members of the public at levels that are ALARA. The concept of BAT is directly concerned with discharge abatement. In particular, since potential new power plants will be based on those currently operating, an understanding of what is achievable for current plants should give valuable information on what may reasonably be expected for discharges from a new plant. Additionally, the use of “best” in BAT implies some notion of comparison as a means of identifying what can practicably be done in terms of reducing discharges to the environment.

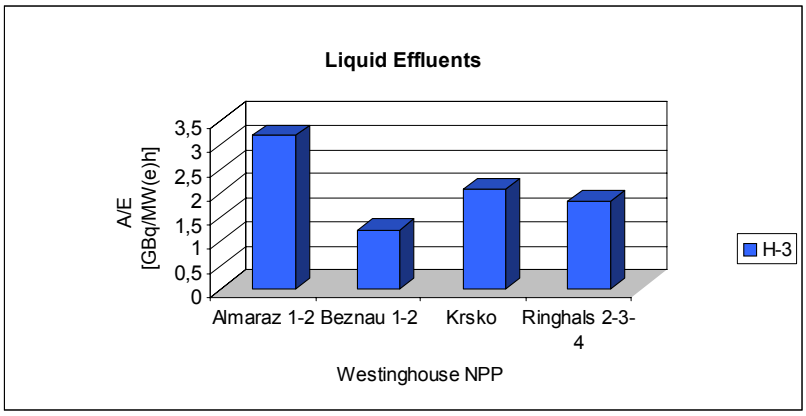
Since member countries record data on discharges from their nuclear power plants, and these data are collated internationally by organisation such as the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), the IAEA and the European Commission (EC), it would seem that drawing conclusions based on discharge data might be straightforward. In fact, currently, although much data is available, it is difficult to use these data in judging what are BAT or optimally managed effluent treatment systems.

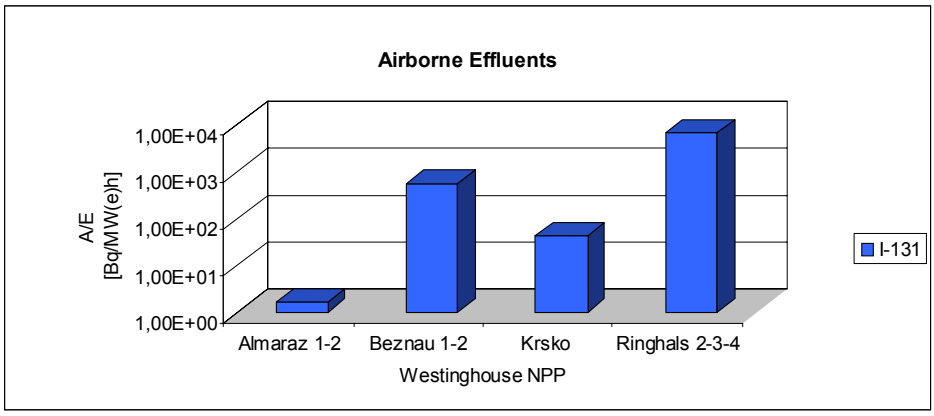
Discharge data from similar plants can vary by several orders of magnitude. It is unclear how far this is due to genuine differences between plants (different BAT or different results of optimisation processes?) and how far it is due to differences between plants (or countries) in monitoring, analysis and reporting of discharge data, which are not harmonised. An example may be the monitoring technique used for a radionuclide such as tritium. There may be differences in results from a monitoring system that is based on continuous sampling rather than a monitoring system that is based on grab sampling, with extrapolation of the results to the remainder of the period for which the results are reported. This may be true even if a plant may have measured data to suggest that the levels of tritium do not vary substantially across a reporting period. It may also be noted, for example, that releases of tritium and carbon-14 may occur in different chemical forms, dependent on reactor type. Harmonisation of measurement and reporting requirements may therefore need to reflect those variations or otherwise account for them.

Therefore, comparison of data on a 'like-for-like' basis at this time may be difficult or even impossible. This situation could be improved by working with available data to allow comparison as far as possible and, in the future, to achieve common basic standards for data recording and reporting. Very broadly (also going beyond the nuclear industry), two trends can be identified in regulation of discharges: first, an integrated holistic approach encouraging an operator to take a global view and, second, a tendency to set more specific expectations, perhaps even for the performance of a particular piece of equipment, to ensure enhanced control over discharges and control of performance. While the detail with which the expectations themselves are specified is perhaps not of concern, they may have an influence on the detail of data collected, which is important.

As an example, Figures 1 and 2 below show discharge data for Pressurised Water Reactors from the same vendor (Radioisotopes in Effluents from PWR NPPs during its Lifetime, H. Janzekovic and M. Krizman, Int. Conf. Nuclear Energy for New Europe, 2006). While the variation in discharge rate is not large for tritium, it can vary by several orders of magnitude for some other radionuclides, as may be seen in Figure 2 for iodine-131.

Figure 1 and 2





Overall, handling (and perhaps availability) of data needs to be improved if it is to be useful in judging the level of performance that may be expected from a new nuclear plant and, thus, identify what are optimised discharge rates or BAT for such plants. In short, it is currently difficult to judge what may be termed, for example, as BAT for new nuclear power plants (of 'Generation 3+') based on similar, currently operating, plants (of 'Generation 3').

To be more specific, OSPAR has stated (in Towards the Radioactive Substances Strategy objectives, third periodic evaluation, OSPAR, 2009) that there is evidence of progress being made in meeting the strategy objectives, but "there are limitations which demonstrate that further work is needed before a future evaluation of progress can be expected to deliver robust overall conclusions". The Radioactive Substances Committee notes that "the presentation of data on discharges from the nuclear sector could be improved, to identify the contributions of exceptional discharges from decommissioning and clean-up and the effects of variability in the level of operation of installations".

In the same OSPAR document mentioned above, the nuclear fuel cycle releases are described in the following manner: "In the nuclear sector, discharge data are collected for four sub-sectors: nuclear fuel production and enrichment, nuclear power plants, nuclear fuel reprocessing and nuclear research. In both the baseline period and the assessment period, the major contributors to discharges were the reprocessing and fuel production and enrichment sub-sectors, with discharges from nuclear power plants and research facilities being relatively small." The intent of reporting this statement is not to suggest that nuclear power plants need expend no more effort in discharge abatement, but rather to point out that in locations where there may be multiple facilities in different states of operation or decommissioning or from various nuclear sub-sectors, it is important to consider how the discharges from a single facility are to be measured and impacts assessed in the more complex situation of the discharges from the multiple facilities.

Another factor to be considered is the handling of discharge data in which some percentage of the results are at values below the detectable levels of discharge given the available instrumentation. World-wide, the handling of such values is not consistent at this time. Under the auspices of OSPAR, a methodology has been defined for potential adoption for the purposes of reporting of releases to the marine environment. That methodology is described in the report titled "Assessment on statistical techniques to the OSPAR Radioactive Substances Strategy", OSPAR, 2009. Three contexts are described:

- A dataset including no "non-detect" values (< DL), for which comparisons of means from the assessment period with the baseline period are done using both parametric and non-parametric statistical tests.

- A dataset including up to 80% of non-detect values, with methods described in the report.

- A dataset including more than 80% of non-detect values. In this case, the data are considered as insufficient and no assessment is performed. Of course, the dataset is still valuable in describing lower and upper bound results for the radionuclide release rates that are the subject of the dataset.

The challenge is to derive the best possible assessment given the available datasets, and to ensure that appropriate conclusions and appropriate methodological limitations are described, including the level of conservatism that may be introduced by the way(s) the non-detect value sets are handled.

6.3 Harmonisation

Considerations for international harmonisation regarding the collection and reporting of discharge data should give due care to the legislative differences in the nuclear countries of the world, since national approaches and priorities may differ. Any work to set basic obligations should build on work already done, given the effort expended to date by such organisations as UNSCEAR, IAEA, and the EC in collating and analysing discharge data from existent facilities of various types. Any basic obligations so developed should focus on enhancing data comparability among facilities of similar

types (e.g., pressurised water reactors, boiling water reactors, and CANDU reactors) so that effective technologies and practices in discharge management may be more easily identified.

OSPAR system of reporting

As one example of an organisation's addressing the potential advantages of further harmonisation in discharge monitoring and reporting, the OSPAR Radioactive Substances Committee has agreed that there should be a set of minimum data reporting requirements for discharges to the aquatic environment. Those reporting guidelines were initially developed in 2004 as "Guidelines for the submission of information on the assessment of the application of BAT in nuclear facilities" (RSC 04/6/1-E). Numerous member states of OSPAR have prepared reports according to those Guidelines. It is reasonable that the guidelines may also be applied to discharges to the atmosphere and to discharge of solid waste (e.g., to the soil). The report format could be modified for those nations and regions for which the BAT concept has not been adopted, but rather for which, the ALARA optimisation process is solely used.

An example of the OSPAR report format for releases of liquid effluent is as follows:

- Sources of liquid effluent
- Liquid effluent treatment and abatement
- Trends in discharge rates over the reporting period (for example, a five-year period)
- Calculated radiological impact of liquid discharges
- The application of BAT at the facility
- Comparison with performance of similar plants world-wide

An extended report format could be developed to incorporate the sources of effluent to the atmosphere and solid wastes to be disposed, effluent treatment and abatement applicable to "gaseous" effluents and solid waste streams, calculated radiological impacts of the discharges to the atmosphere and for solid waste disposal, the application of BAT for the effluent to the atmosphere and for solid waste disposal, and comparison of performance of similar plants world-side. To ensure that an integrated view is taken, the application of BAT for the total discharge abatement process could be evaluated, as results of such an evaluation may differ from those of reviews taken on solitary effluent streams.

For an operating company hosting several similar facilities, the company may compile the data from singular facilities into a report drawing conclusions for the several facilities. For a country hosting several similar facilities, the regulator or other appropriate authority may compile the data from singular facilities into a report drawing conclusions for the several facilities. As an example, a country may report on fuel manufacturing facilities of a similar type, power generation facilities of one or more similar types, research and development facilities of a similar type, radioisotope manufacturing facilities of a similar type, and radioisotope use facilities (e.g., medical and radiographic facilities) of one or more similar types.

6.4 Discharge experience from power upgrades

The thermal power of the Loviisa (PWR) and Olkiluoto (BWR) nuclear power plants (each having two units) has been upgraded. No effect of the power upgrades on the discharges of fission products and activated corrosion products has been detected. Other factors than relatively small power upgrades apparently affect discharge rates more significantly. However, increase in the discharges of H-3 and C-14 has been clearly noticed. Their production depends on the energy production and almost all is discharged into the air and sea for these two reactor sites.

Sizewell B Power Station was up-rated to 101% of Rated Thermal Power in June 2005. The consequent increased boron concentration at the beginning of the fuel cycle nominally increased the amount of tritium produced from this source term. There were similar percentage increases from those other tritium sources that are power dependent. However, these effects were not noticeable in practice, as the rate of change of tritium production was dominated by increases in tritium production from Secondary Source Assemblies (SSA) that had been replaced during the previous Refuelling Outage. These SSA take up to one-thousand days to reach equilibrium. The total tritium production for the fuel cycle was within the bounds of that calculated without taking account of the power up-rate. There have been no other measurable effects of the power up-rate.

In some countries of the world, power up-rates of nuclear power plants on the order of 10-20% have been authorised and implemented. Because the generation of radioactive materials is for some radionuclides directly proportional to the operating thermal power of a reactor, it is reasonable to presume that for some radioactive materials subject to discharge to the environment, power up-rates would (without at least equivalent improvement in discharge abatement results) result in increased discharges to the environment. Such increases may not be directly detected via discharge monitoring techniques, depending on the actual level of increase of the discharge and the limits of detection of the discharge monitoring techniques. As operating companies request authorisations for power up-rates and as regulators review such requests, there should be written evaluation of the anticipated increases in discharges to the environment and any offset or abatement of those increases that may be expected due to reasonably available improvements in discharge treatment or other systems in the facility. When the up-rates have been implemented, the operator should include discussion of any actual increases detected in releases to the environment and whether techniques are available to the operator to offset those increases.

7. USE OF COLLECTIVE DOSE (IN OPTIMISATION AND IN BAT ASSESSMENT)

There are two measurements of detriment in terms of dose that are commonly recognized. The first is dose to an individual and the second is collective dose to a group of individuals. This section of the report addresses primarily the concepts of collective dose and when and how such dose is likely to be used in assessment of ALARA and BAT. For purposes of comparing potential protective actions in the optimisation process, a way to characterise individual dose distributions within groups is the use of the collective dose associated with those distributions.

As protection is optimised via a process related to the radiation source being evaluated, a form of cost-benefit (or multi-attribute) analysis may be used in deciding what is reasonably available or reasonably achievable in terms of technology introduction or management practice refinement. In such cost-benefit analyses, the trade-off is usually between the reduced levels of individual dose (and/or reduced level of collective dose) if the activity is performed and the increment in cost to perform the activity. The comparison for a single potential protective action is thus between a change (reduction) in net detriment (measured in terms of collective dose) and a change (increase) in net cost in performing the activity. In some countries and for some situations, a monetary value for a unit of collective dose reduction has been established. For those countries and situations having established such a value and for those countries and situations for which such a value has not been established, it is clear that there are difficulties in establishing such a value because of the uncertainties and limitations inherent in the process. Part of the reason for those difficulties is that judgements are not necessarily inherently quantitative but are complex because of the somewhat subjective social judgements that may be involved.

There may be a level of dose (risk) that has been evaluated to be low enough to be optimised in some countries or situations. (For example, the cost of regulating a source is determined to exceed costs associated with a reasonable conservative estimate of any radiation detriment from such a source.) If the dose (risk) is at or below the level evaluated to be optimised, then in those countries or situations, there is in general no requirement for further optimisation evaluation related to the radiation source. (Changed technical, economic, legal or social contexts that would warrant re-evaluation of the determination should be considered.) For other countries or situations, even though a very low dose may be projected, further optimisation related to that radiation source may be appropriate to evaluate but may also be unlikely, as costs for further dose reduction are more likely to exceed the benefits of whatever further dose reduction may be realistically feasible. In terms of assessing if BAT are being applied, one question still remains applicable. That is, is it reasonably feasible to eliminate the use of the radiation source entirely and use BAT for which overall risk to the environment is improved because the radiation source is no longer present (and presumably, no other source of risk to the environment has been introduced)? Absent an affirmative answer to that question, then for those activities for which no additional options deemed cost-beneficial have been identified, optimisation has occurred, and BAT are by definition in place.

Defining sources for which dose (risk) has been pre-evaluated to be optimised may be established by a country's using exclusion or exemption processes as defined by the ICRP and implemented in the IAEA's International Basic Safety Standards. Those processes relate to a determination that use of such a source results in small individual doses and small collective doses (with costs of control

exceeding costs of reasonably estimated radiation detriment) or to a determination that no reasonable control procedures can achieve significant reductions in individual or collective dose. (The costs of operator and regulatory agency efforts to assess the risk are one component of the costs to be considered in assessing the need for change in technology, practice, or regulation.) The ICRP suggested (in its Publication 64) that small individual doses are those on the order of 10 uSv per year and small collective doses are those on the order of 1 person-Sv per year. In its Publication 101, the ICRP clarified its position that optimisation is to be continued, regardless of projected doses, if options are identified for which the benefits of implementation exceed the costs of that implementation.

The concept of collective dose can be useful, but also is subject to potential misuse. The ICRP has asserted (in Publication 101) that "the collective dose may be a useful input to the optimisation process when the individual dose distributions are relatively homogenous and well defined". In evaluating multiple potential protective actions, the potential actions may be viewed in terms of the absolute value of calculated dose avoided by implementation of an option at the projected cost, but may also (and preferentially) be viewed in terms of the relative value of the implementation of one option as compared to implementation of another option, both at their respective projected costs for implementation. (That is, the relative differences between collective doses avoided by implementing the protective actions being considered would be of importance if choosing between options.) Comparisons should, of course, be made regarding protective actions that have been analysed to be cost-effective to implement, with those options which have been analysed to not be cost-effective having been eliminated from further consideration.

The potential for misuse generally arises when additions are made over a wide range of individual doses, over very long periods of time, or over large geographical areas, and that summed dose is multiplied by a value asserted to be related to total detriment or even health effects. The difficulties are that the dose assessment process includes inherent uncertainties and also that there is insufficient knowledge of the risk coefficients across such ranges of individual dose and over time and over space. Given for example the large degree of uncertainty of detriment at very low source-related levels of exposure (in an environment of larger exposures to natural sources of exposure), the use of the collective dose concept in such a circumstance is not likely to be valid. As an additional example, assessment of collective dose into the far future is considered by the IAEA (in Safety Guide WS-G-2.3) to be "highly speculative" and "may invalidate the results of the analysis".

The ICRP has suggested in its Publication 101, one methodology of dose assessment that may increase the potential for having a tool that is useful in decision-making. A collective dose including a large range of individual doses may be broken up into a series of collective doses that would include reasonably homogenous parts of the overall dose distribution. For example, use of a matrix consisting of collective doses segregated by the spatial distribution of the exposed population, the time distribution of the population, and the individual dose distribution of the population, may be appropriate. In its Publication 64, the ICRP noted that if a dose distribution is developed that covers no more than two-three orders of magnitude (and knowledge is available of population size, mean individual dose, and uncertainty for that distribution), the risk assessment should state that the most likely number of excess health effects is zero when the collective dose is smaller than the reciprocal of the risk detriment. In its Publication 101, the figures stated above do not appear, but a description of a weighting process is included that would reflect the degree of uncertainty in the level of exposure and other factors that may be relevant. For example, a weighting scheme may give more importance to groups receiving higher doses relative to other groups and more importance to groups receiving doses in the next few generations relative to other groups who might be projected to receive doses in the more distant future. The ICRP recommends that predicted doses beyond the next few generations of individuals "should not play a major part in decision making". The Commission also states the

decision on how to treat uncertainties in estimation of dose for compliance purposes should be made by the relevant regulatory authority.

8. CONTINUAL IMPROVEMENT APPROACHES FOR OPTIMISATION AND BAT

Implicit in the optimisation process and in the use of 'available' in BAT is the idea that what is optimum (ALARA) or BAT may change with time as new techniques (technologies and practices) are developed to a point where they may be deployed on an economically feasible scale. The point at which a technology (or practice) becomes the optimum or best available technology (or practice) may depend on the plant in question, chiefly because it is generally cheaper to incorporate a technology (or practice which may include a technological component) into a new plant than to 'back-fit' it into an existent plant. (It might become feasible at a later date for an existent plant, for example, as the technology is refined.)

Although availability of new or improved technology is clearly a key reason for continual improvement, other factors can be at least as important as drivers for adopting new techniques. An example may be enhanced harmonisation in terms of using a particular technological means of monitoring discharge rates. Another example be the role of socio-political drivers in, for example, a nation's adopting policies or non-binding international agreements, or responding to pressure toward a particular end (e.g., in effect, changing what economically or reasonably "available" means by increasing the priority of discharge abatement relative to a factor such as the cost of electricity). As already noted, foremost amongst these is the apparent general approach to radioactive waste management via concentrating and containing wastes when practicably feasible and not offset by increases in doses to workers.

By way of a concrete example, the description below provides a case study illustrating adoption of a new best available technology at the Sellafield reprocessing complex in the UK.

Case Study: Abatement of 99Tc discharges from Sellafield

At Sellafield, the operation of the Magnox fuel reprocessing plant creates a number of liquid waste streams which are evaporated to form Medium Active Concentrates (MAC). MAC contain, inter alia, technetium-99 and actinides such as plutonium and americium.

Prior to 1981, MAC were discharged to the sea after several years' storage to allow for decay of short-lived radionuclides. In the early 1980s, discharges of MAC were suspended and the concentrate was retained in storage, pending the construction and operation of the Enhanced Actinide Removal Plant (EARP). While EARP was designed primarily to remove alpha-emitting radionuclides, a number of beta-emitting radionuclides are also removed. However, one exception was Tc-99 which was not separated in the EARP process.

EARP began operation in 1994 and started to clear the backlog of stored waste, as well as ongoing materials input from Magnox reprocessing. Consequently, there was a significant increase in the discharge of Tc-99 to sea. This technetium was detectable in the environment of the Irish Sea and in certain marine species, including shell-fish, and over time, became detectable outside of UK waters. The environmental concentrations of technetium received a large amount of national and international attention.

In February 2000 the UK's Environment Agency (EA) initiated a full re-examination of authorisations for the disposal of radioactive wastes from Sellafield. This included a programme of research and development to determine the best practicable means for reducing Tc-99 discharges. The review covered characteristics such as abatement options (including a detailed review of available technologies), potential process changes, impact assessment, storage options and costs.

Following the review, the EA directed British Nuclear Fuels Ltd. (BNFL) to (i) re-route future materials input of MAC, from the reprocessing of Magnox fuel, to the Highly Active Liquor Evaporation and Storage Plant, for subsequent vitrification and (ii) to investigate the potential to use TPP (tetraphenylphosphonium bromide) to remove Tc-99 from the historic/stored MAC.

BNFL implemented the first of the above requirements (MAC diversion) in July 2003. With regard to the second requirement, significant concerns about the management and disposability of the solid waste product needed to be addressed (via a research programme).

It is noteworthy that a study carried out by the site operator, BNFL, concluded that "there is negligible probability of implementing a TPP process in EARP/WPEP that captures technicium in an oxidation state and in a waste form acceptable to the current Nirex safety case" and recommended that the Best Practicable Environmental Option for Tc-99 was its continued discharge to sea. However, following close collaborative work between BNFL, the EA, and the Nuclear Installation Inspectorate (NII) and Nirex, a trial of TPP in EARP was carried out in October-November 2003. The trial was a success and since April 2004 the new technique has been in use at EARP.

Over 95% of the Tc-99 contained within the stored MAC has been transferred into a solid waste form for encapsulation. This has led to an overall reduction of about 90% in Tc-99 discharges from the site to the sea.

9. ASSESSMENT FOR FACILITY DESIGN THROUGH FACILITY DECOMMISSIONING

An overarching aim of environmental (including human) protection is reduction of the potential for an overall adverse impact from an activity, such as producing electricity from a nuclear power plant. The use of BAT, as applied to routine discharges, is focussed on just one aspect of a plant's impact. To optimise the plant's performance overall, a holistic view needs to be taken, beginning with the design and construction of the plant, to effective management of radioactive and other substances, to consideration of effects on waste streams from the plant (and consequent potential impact on workers, the public and the environment).

In the past each stakeholder (or specialist field) tended to focus on one specific well-defined goal; rather, a holistic "design through decommissioning" (sometimes somewhat inappropriately called a "cradle to grave") approach would integrate the goals and perspectives of all relevant stakeholders (and specialist fields) and reduce the possibility of generating internal conflicts between different stakeholders. For example, procedures for treatment of a radioactive waste at a plant should harmonize with the goals of the radioactive waste management organisation which will eventually be responsible for the waste. Transport may also be an important factor, potentially affecting the choice between on-site and centralised waste processing facilities.

In addition, a holistic design approach considering the lifecycle of the facility should take into account that requirements related to radiation safety standards and other standards can be analysed early in the process and design or programme enhancements perhaps can be implemented on a site at little or no additional cost early in the process. Standards used in the chemical industry should be taken into account as well; this aspect is discussed in more detail elsewhere in this document.

It is expected that new nuclear power plants will have a long operating lifetime, in the range of 60-80 years, requiring substantial pre-planning for age-related impacts. Over such a period of time not only will personnel change (implying challenges in knowledge management) but also the technical characteristics of some of the equipment at a plant could drastically change. An integrated lifecycle design approach should take into account this fact so that all documentation related to the site design would be properly preserved and flexibility designed into the structure of the facility as practicable, to facilitate equipment upgrading or replacement.

Last but not least the need for the lifecycle approach to design is based also on the fact that solutions related to effluents, radioactive waste and decommissioning which were developed in the past were upgraded multiple times, to satisfy the changing expectations of the various stakeholders and to ensure there were no gaps or discontinuities going from one lifecycle phase to another.

A possible approach to assist in achieving an integrated, lifecycle approach is given in the text box below.

A lifecycle approach through stakeholder involvement

The lifecycle approach requires a regulatory authority as well as an operator to analyse all phases of the lifecycle of a facility in relation to the stakeholders involved in its safe operation. The techniques could be described by two steps:

1. A role for the various stakeholders, namely designers (vendors), constructors, technical support organisations, maintenance organisations, waste management organisations, international organisations as well as regulatory authorities and others, should be defined for each phase of the lifecycle. A list of stakeholders could be even larger in specific cases, for example when transport of nuclear fuel takes place. In such cases also transporters, customs authorities or local law enforcement personnel could be recognised as stakeholders.

2. In each phase of the lifecycle, radiation safety issues should be recognised and goals should be set. The goals could be set for example in a frame of performance indicators as for example the volume of low level waste or the annual release of C-14 in airborne effluent from a PWR. The influence of each stakeholder on a specific goal or performance indicator should be recognised in advance.

Using this technique, an operator of a facility as well as the relevant regulatory authority could optimise its manpower as well as its financial resources to achieve a prescribed goal in the appropriate timeframe. The techniques require upgrading from time to time as new technical developments or scientific facts are demonstrated to be applicable. Also, the managerial issues could require redefinitions of roles over time.

9.1 National approaches to LLW and ILW storage and disposal

The Loviisa and Olkiluoto power plants both have their own facilities for intermediate storage and final disposal of low level radioactive wastes and intermediate level radioactive wastes (LLW/ILW). They have storage buildings for intermediate storage of LLW/ILW. The waste is disposed of in bedrock 60 - 100 m below the ground surface. Very low level waste can with appropriate evaluation be released from regulatory control or alternatively, buried in a disposal facility authorised for the specific purpose.

9.2 National solid waste disposal criteria will affect BAT choices

National solid waste disposal criteria don't affect on BAT choices in Finland, in practice, at the moment. An issue could be the final disposal of C-14 (dose criteria) but at the moment almost all C-14 is discharged to the air.

For Sizewell B, the UK criteria have conditioned the types of discharges that they produce.

<u>Very Low Level Waste (VLLW)</u>

< 4 MBq te^{-1} total activity or < 40 MBq te^{-1} of tritium

May be disposed of to a specified landfill site subject to Authorisation.

Low Level Waste (LLW)

Radioactive Waste having a radioactivity content not exceeding 4 GBq te^{-1} of alpha emitting nuclides or 12 GBq te^{-1} of beta/gamma emitting nuclides, regardless of half-life or radiotoxicity.

May be disposed of to the Low Level Waste Repository (LLWR) in Cumbria or to an Authorised disposal facility, for example an Authorised waste incinerator.

Intermediate Level Waste (ILW)

Radioactive Waste having a radioactivity content exceeding 4 GBq te^{-1} of alpha emitting nuclides or 12 GBq te^{-1} of beta/gamma emitting nuclides, but not generating heat.

No disposal route in the UK

Long term storage at designated sites.

High Level Waste (HLW)

Heat generating radioactive waste, usually associated with nuclear fuel.

No disposal route in the UK.

Long term storage at designated sites.

When treating liquid and gaseous wastes, consideration must be given as to whether to generate greater volumes of LLW or smaller volumes of ILW. Currently there is no disposal route for ILW or HLW in the UK, so that the need to store such waste in a passively safe state adds weight to operating regimes that minimise production of ILW.

At Sizewell B, it is recognised that treatment of Reactor Coolant, in the Chemical and Volume Control System, and continuous cooling and clean-up of the Fuel Storage Ponds will generate ILW filters and demineraliser resins. Other potential sources of ILW are filters from the Resin Transfer System, decontamination wastes and concentrates from the evaporator systems. The lack of a suitable treatment and disposal route for evaporator concentrates, coupled to the expected operator doses and technical difficulties operating and maintaining the evaporator systems, whilst delivering marginal public dose benefits, led to the decision during commissioning not to use evaporation for the treatment of liquid wastes, so avoiding the evaporator concentrates waste stream from being created. This is an example of how national policy on solid waste management affects BAT choices.

10. CONCLUSIONS

A wave of new construction of nuclear power plants is anticipated to occur in various countries and regions of the world. The general tendency to move away from a "dilute and disperse" towards a "concentrate and contain" approach to radioactive waste management means that the level of discharges from these plants will be a key factor in their broad acceptability and important for their regulation and operation.

Increased consolidation of reactor vendors, and the consequent implication that nearly identical plants will be widely deployed, means that some level of harmonisation around what is the best reasonably achievable discharge performance is desirable, both to avoid repetition of work and because the increasing similarity of reactor designs may lead to increased expectations of comparable performances.

Discharge abatement is based on a trade-off between costs and benefits, through reducing doses (application of optimisation via the "ALARA" principle) or through adoption of optimisation via the use of Best Available Techniques (or their equivalent), for reducing discharges. A general understanding of BAT recognises the importance of economic and social factors in deciding the technique (technology and practices) of choice, notably that technique deployment must be feasible.

The principles of maintaining doses to members of the public at levels which are as low as reasonably achievable and the principles of using "best available techniques" to control releases of radioactive materials to the environment are both approaches to optimisation and are complementary to one another. Both approaches are compatible in their objectives of limiting doses (risks) to humans, foreseeable effects on non-human species, and releases of radioactive materials to the environment. Both approaches are implemented via structured, on-going, forward-looking processes that evaluate reasonably available technological and process/procedure options for management of discharges of radioactive materials to the environment. Application of the optimisation principle, to achieve exposures that are as low as reasonably achievable (ALARA), is focused more explicitly on ensuring doses to individuals from a source are appropriately controlled, while application of the BAT principle is focused more explicitly at ensuring that discharges from that source are appropriately controlled. The concepts of achieving exposures that are ALARA, and of using BAT to control effluents, are complementary approaches to optimisation. The phrase "best available techniques" tends to be used more often in western Europe, whereas the term "optimisation" is used more globally and may be in some regions more often discussed using the terms "as low as reasonably achievable" or "as low as reasonably practicable".

In further understanding of the complementary nature of the ALARA and BAT approaches, several issues to be addressed were identified:

- Establishing a mechanism to define which technologies may be confidently applied to the nuclear sector in a technically and economically viable manner. The current approach taken for BAT is Eurocentric, and additional clarity is needed regarding effective means to apply the methodology across regions of the world.

- Different results may be forthcoming in evaluations using factors defined to include value judgements regarding terms such as "reasonable", "unreasonable", "excessive", and "grossly

disproportionate". Discussion regarding the value judgements and their bases and interpretation may be appropriate in the various national (and perhaps international) settings as changes to regulations are considered.

- There are ongoing scientific evaluations as to how non-human species and the environment overall are to be protected. Industry, regulatory, and other stakeholders may be desirous of discussing whether adoption of a BAT concept is timely or potentially may be premature, as results of those evaluations become available over the next several years.

- Relating to the first and second matters listed, there may be a need to establish clarity regarding the level of effort required of the operator to attempt to identify potentially feasible means of reducing discharge rates, whether optimisation is by some considered effective only when the result is an ongoing measurable reduction in discharge rates, how the balance between nuclear safety, industrial safety, radiation safety, and environmental safety is measured and judged, and how the balance between reduced power generation and reduced discharge rates is judged.

The application of BAT for radioactive waste discharges is not to be done in isolation from other aspects of nuclear power plant operation. BAT should be identified taking a holistic view across the many considerations affecting plant operation and maintenance of plant safety (nuclear, radiological, environmental, and industrial). Bearing in mind this broad view of implementing BAT, the following key points were identified in considering implementation of the BAT concept by an operator and/or regulator:

- Common understanding of BAT for a specific facility

- Definition of optimised radioactive waste discharge rates for the facility

- Establishment by the regulator of authorised discharge rates for the facility

- Discharge data and their comparability to data from similar facilities

- Continual improvement of BAT as technologies and operating practices are refined and shown to be reasonably feasible for the application

- The possibility of international harmonisation of BAT implementation approaches

- An integrated view: assessment applicable from facility design through facility decommissioning, considering radiation, chemicals and other potential risk agents

- BAT for emergencies and incidents and the relationship to the nuclear safety culture

- Management systems in establishing and maintaining BAT implementation

- Ancillary strategic issues: loss of expertise and the impact of the market place.

The most important topic that can usefully be addressed when considering BAT for new nuclear power stations is analysis of current discharge data to arrive at a clear understanding of what may reasonably be expected from new plants, which will be evolutions of current designs.

Annex 1

BAT for Facilities Outside of the Nuclear Fuel Cycle

There are many facilities world-wide using radioactive materials in their processes or containing radioactive materials in their raw materials or their output/product. These include facilities of the nuclear fuel cycle but also many other types of facilities. Examples may be medical diagnostic and treatment facilities, research and development facilities, and mineral extraction facilities.

Some of these facilities use radioactive materials only as contained or sealed sources, such that the only detectable potential impact on man or biota is via the direct radiation being emitted from the source used (until disposal of the decommissioned source). Any exposure to a member of the public decreases very quickly with distance from the source, and the direct radiation often is at levels detectable only very close to the source.

For other facilities, radioactive material may be used or conveyed that can result in discharges to the air or water or as solid waste. The radioactive material may be a radionuclide such as iodine-131 for nuclear medicine treatment of an individual member of the public. The radioactive material may be a radionuclide such as sulphur-35 used as a tracer in a university biomedical research project. Or, the radioactive material may be radionuclides within a naturally occurring nuclide series that co-exists with a mineral being extracted for beneficial use. For all such facilities, similar discharge monitoring, environmental monitoring, and reporting of applicable data may be both prudent and required. Further, similar questions apply regarding whether BAT have been implemented at the facilities. The level of investigation regarding if there may be practicable means to further reduce discharges may be partially dependent on the potential impact of the discharges on man and biota. That is, for facilities with very small discharges and very low calculated environmental impact, the operator and regulator may be expected to expend fewer resources on the repetitive evaluations of BAT implementation than for those facilities with larger discharges and a higher potential environmental impact.

The amount of detailed data available on discharge rates of radionuclides or groups of radionuclides from some non-nuclear-fuel-cycle facilities is not sufficient (at least over a long enough time period) to draw firm conclusions about the practicality of reducing discharge rates. This is a matter for the facility operators and the appropriate regulators to address. Some tentative findings appear to be the following:

- Offshore oil and gas facilities release thorium and uranium series nuclides to the environment in effluent pathways such as produced water and (with a small contribution from the) disposal of wastes from descaling operations. For the signatories of the OSPAR convention addressing releases to the marine environment, a very substantial percentage of total alpha discharges from all sectors and a not-insubstantial percentage of total beta discharges from all sectors comes from the offshore oil and gas facilities. Caution should be exercised when making comparisons, in particular due to differences in measurement and calculation techniques. For example, in the off shore oil and gas sector, total alpha and total beta discharges from produced water are calculated using the contributions from the radioactive daughter products in the respective uranium/thorium decay chains. The formulae assume equilibrium in these decay chains, and consequently the calculated total alpha and total beta values are the maximum activities that can be produced from the decay of the measured/assessed parent radionuclides (i.e. Ra-226, Ra-228 and Pb-210). Thus, the resulting total alpha/beta discharges are the calculated upper bound of activities discharged, rather than a measured total alpha or total beta discharge (as is the case for values reported for the nuclear sector).

- For the OSPAR signatory parties, iodine-131 discharges from the medical sub-sector are the single largest contributor to total beta discharges (excluding discharges of tritium) from all sectors.

- For the OSPAR signatories, university, research and development, radiochemical manufacturing and other facilities appear, from the very limited data available to the reviewers, to be of minor importance.

Regarding further evaluation of the sectors outside of the nuclear fuel cycle, the following considerations are relevant:

The operators and regulators of offshore oil and gas facilities with radioactive material discharges should continue to gather, report, and assess data on alpha, beta, and where appropriate, radionuclide-specific discharge rates from the facilities. Comparison to discharge rates from similar facilities is appropriate, to evaluate differences for generation of appropriate discharge-abatement methods. Since 2005, OSPAR contracting parties (CP) report annually the estimated discharges from offshore installations of radioactive substances: (i) in produced water (Pb-210, Ra-226, Ra-228), (ii) from descaling and decommissioning operations (Pb-210, Ra-226, Ra-228, Th-228), and (iii) from tracer experiments (H-3, other beta and gamma emitters). The reported data will be subject to a thorough assessment when sufficient data is available. It should be noted however that the discharge data are reported by each CP for their entire offshore oil and gas sector and not for individual facilities.

The operators and regulators of medical sub-sector facilities should ensure the reporting and assessment of iodine-131 discharge rates. Should comparison to similar facilities suggest that technetium-99 discharge rates are substantially higher than for similar facilities, those discharge rates should also be considered in the reporting and assessment process. In 2005, OSPAR CPs commenced annual reporting of (estimated) iodine-131 and technetium-99 (arising from the decay of the medical product technetium-99m) discharges from the medical sub-sector. In 2009, it was decided that the data collection for technetium-99 was no longer required as very little technetium-99 was generated from the medical use of technetium-99m and consequently no data was provided for 2008. In addition, reporting of iodine-131 discharges is not required where delay tanks are used to deal with liquid effluents.

- For radiochemical manufacturing or other facilities releasing tritium-labelled organic compounds, the operators and regulators may wish to evaluate the setting of discharge-rate thresholds above which the availability of reliable effluent and environmental monitoring data should be ensured. Tritium discharges from the radiochemical manufacturing sector are reported annually to OSPAR. These discharges, while substantially lower than for the nuclear sector, contain a proportion in the form of tritium labelled organic compounds.

- Operators and regulators should continue to review if there are reasons to believe that discharges from any of the sectors listed below may be increasing relative to those of the nuclear fuel cycle, offshore oil and gas facilities, or nuclear-medicine treatment facilities:

  - university and research

  - mineral extraction, e.g., phosphate

  - titanium dioxide pigment manufacturing

  - primary steel manufacturing

- manufacture of gaseous tritium light devices or ionising-chamber smoke detectors.

OSPAR has been collecting data for discharges of radioactive substances from the non-nuclear sector only since 2005 and hence the amount of data available for this sector is limited. OSPAR will assess the time trend in these discharges, and where applicable marine concentration data, as further information becomes available.

References applicable to the discussion above and for additional information are the following:

Discharges of radioactive substances from the non-nuclear sectors in 2007, OSPAR Commission, 2009.

Towards the Radioactive Substances Strategy objectives, Third Periodic Evaluation, OSPAR Commission, 2009.

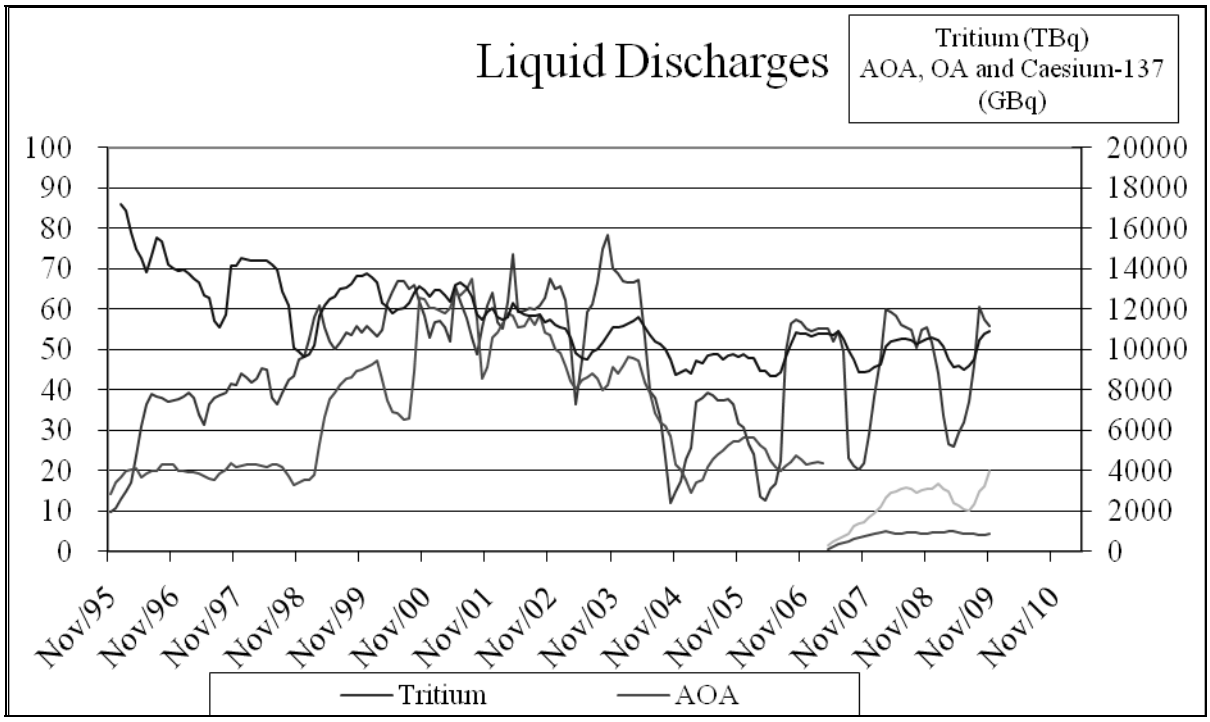
Additional publication:

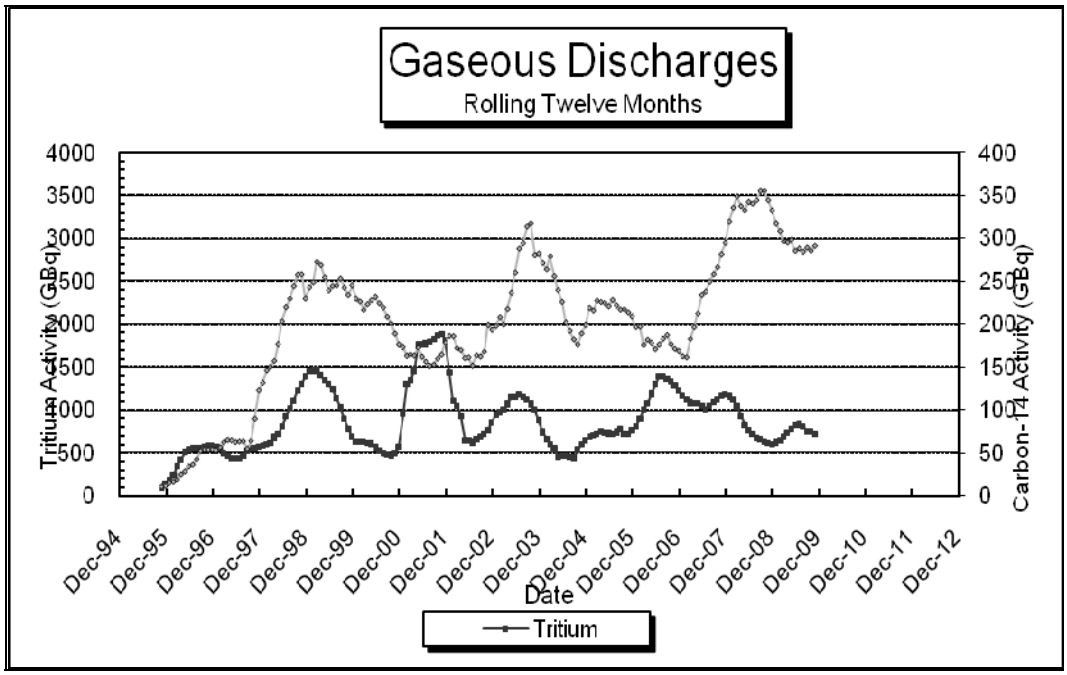
Discharges of radioactive substances from the non-nuclear sectors in 2008, OSPAR Commission, 2010.

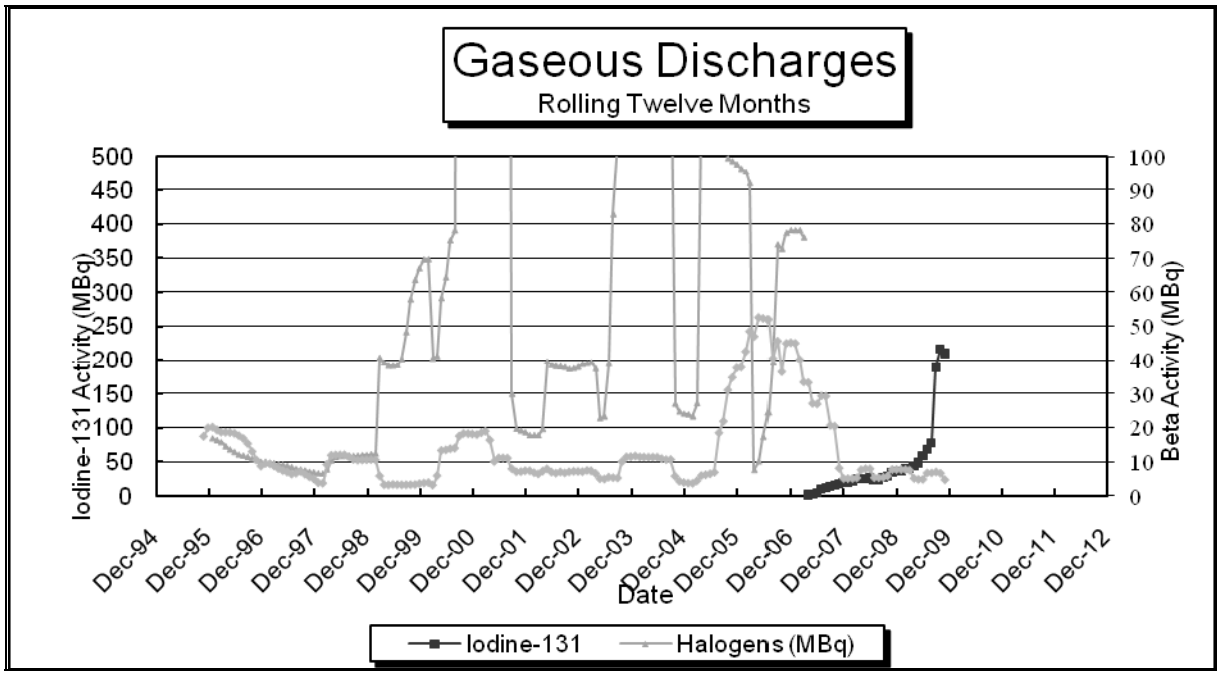
Annex 2

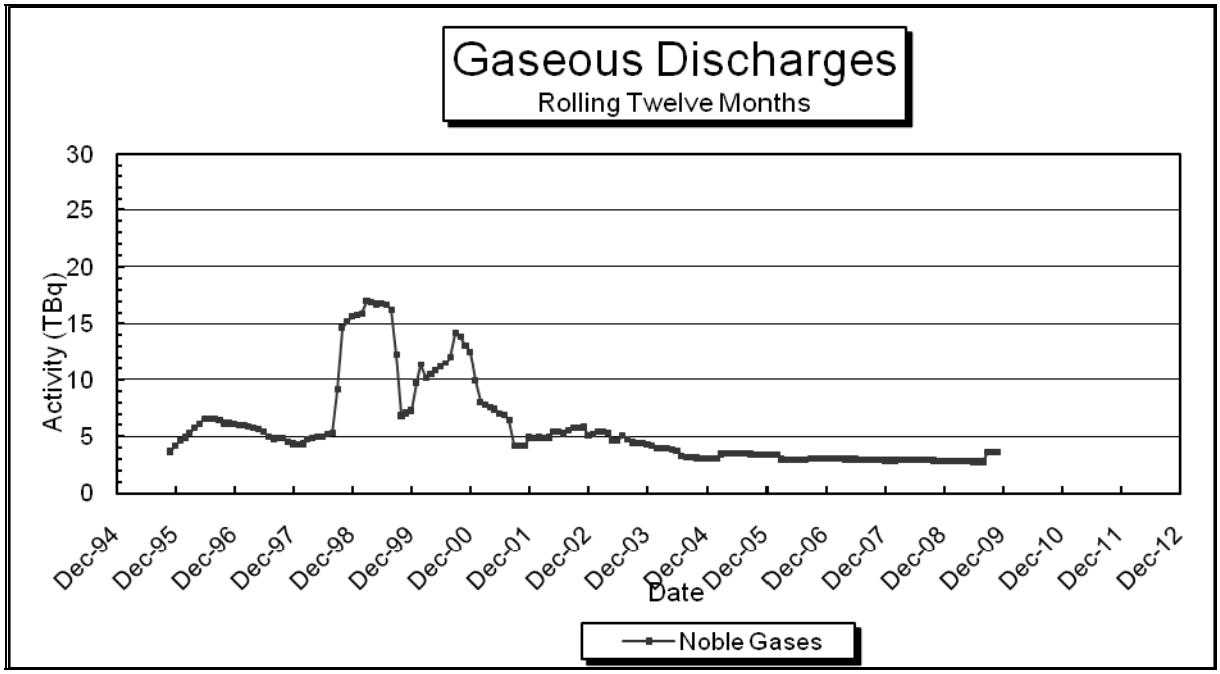
Representative Discharge Data

(supplied by Sizewell B)






Gaseous Discharges
Rolling Twelve Months
Iodine-131 Activity (MBq)
Beta Activity (MBq)
Date
Iodine-131
Halogens (MBq)


Gaseous Discharges
Rolling Twelve Months
Activity (TBq)
Date
Noble Gases

Annex 3

Representative Discharge Data from Finland

Annual Report of Finland, 2008

Radionuclide discharges from nuclear power reactors during normal operation

Reported values are based on measurement results. Commission recommendation on standardised information on radioactive airborne and liquid discharges into the environment from nuclear power reactors and reprocessing plants in normal operation (18 December 2003) is not fully followed. All required data are not available.

Loviisa NPP's laboratory discharges are included in liquid discharges of Table II.

Table I	Airborne discharges from Loviisa NPP (PWR)
Table II	Liquid discharges from Loviisa NPP (PWR)
Table III	Airborne discharges from Olkiluoto NPP (BWR)
Table IV	Liquid discharges from Olkiluoto NPP (BWR)

Table I

Airborne discharges from nuclear power reactor	
Reactor site (name/type): Loviisa NPP, PWR	**Period (year of discharge):** 2008
Air volume released during the period (m^3): 2,6E+09 m^3	

Category/Radionuclide	Highest value of detection limit actually achieved for key nuclides (Bq/m^3)	Activity discharged per year (Bq)	Commentary
Noble gases			
Ar-41		3,63E+12	
Kr-85	**8,0E+03**	**0**	
Kr-85m		2,47E+10	
Kr-87		2,23E10	
Kr-88		5,67E+10	
Kr-89			
Xe-131m			
Xe-133	**1,0E+02**	**1,58E+11**	
Xe-133m			
Xe-135		1,39E+11	
Xe-135m			
Xe-137			
Xe-138			
Sulphur-35			
Particulates, excluding iodines			
Cr-51		7,88E+06	
Mn-54		1,00E+06	
Co-58		4,19E+06	
Fe-59		1,99E+05	
Co-60	**6,0E-04**	**1,60E+06**	
Zn-65			
As-76		2,71E+07	
Sr-89		3,21E+04	
Sr-90	**1,0E-04**	**0**	
Zr-95		4,74E+05	
Nb-95		1,15E+06	
Ag-110m		4,30E+06	
Sb-122		1,03E+06	
Sb-124		4,61E+06	
Sb-125			
Te-123m		3,13E+05	
Cs-134			
Cs-137	**5,0E-04**	**1,10E+04**	
Ba-140			
La-140			
Ce-141			
Ce-144			
Pu-238			
Pu-239+Pu-240			**MDAs are not reported by NPP, total alpha is measured**
Am-241			**MDAs are not reported by NPP, total alpha is measured**
Cm-242			
Cm-243			
Cm-244			
Total alpha		8,37E+04	
Iodines			
I-131	**2,0E-03**	**1,58E+06**	
I-132		4,56E+06	
I-133		1,66E+06	
I-134			
I-135			
Tritium	**1,0**	**2,71E+11**	
Carbon-14		3,29E+11	

Table II

Liquid discharges from nuclear power reactor	
Reactor site (name/type): Loviisa NPP, PWR	**Period (year of discharge):** 2008
Water volume released during the period (m^3): 1,9E+04 m^3	

Category/Radionuclide	Highest value of detection limit actually achieved for key nuclides (Bq/m^3)	Activity discharged per year (Bq)	Commentary
Other nuclides (excluding H-3)			
Be-7			
S-35			**MDAs are not reported by NPP**
Na-24		3,62E+05	
K-42		7,13E+07	
Cr-51		1,53E+07	
Mn-54		1,00E+07	
Fe-55			
Fe-59		2,00E+06	
Co-58		2,58E+07	
Co-60	**1,0E+03**	**2,80E+07**	
Ni-63			
Zn-65			
As-76			
Sr-89			
Sr-90	**2,0E+02**	**0**	
Zr-95		4,24E+06	
Nb-95		5,93E+06	
Nb-97			
Mo-99			
Ru-103			
Ru-106			
Ag-108m			
Ag-110m		6,06E+07	
Sb-122		5,68E+06	
Te-123m		3,73E+06	
Sb-124		4,92E+07	
Sb-125			
I-131		4,90E+05	
I-133		1,02E+05	
Cs-134		8,05E+05	
Cs-137	**1,0E+03**	**6,62E+06**	
W-187			
Ba-140			
La-140			
Ce-141			
Ce-144			
Pu-238			
Pu-239+Pu-240			**MDAs are not reported by NPP**
Am-241			**MDAs are not reported by NPP**
Cm-242			
Cm-243			
Cm-244			
Total alpha			
Tritium		1,71E+13	

Note: Loviisa NPP's laboratory discharges are included in liquid discharges.

Table III

Airborne discharges from nuclear power reactor	
Reactor site (name/type): Olkiluoto NPP, BWR	**Period (year of discharge):** 2008
Air volume released during the period (m^3): 6,2E+09 m^3	

Category/Radionuclide	Highest value of detection limit actually achieved for key nuclides (Bq/m^3)	Activity discharged per year (Bq)	Commentary
Noble gases			
Ar-41			
Kr-85	**-**	**–**	**not measured**
Kr-85m			
Kr-87			
Kr-88			
Kr-89			
Xe-131m			
Xe-133	**5,3E+02**	**0**	
Xe-133m			
Xe-135			
Xe-135m			
Xe-137			
Xe-138			
Sulphur-35			
Particulates, excluding iodines			
Cr-51			
Mn-54		2,98E+06	
Co-58		1,15E+06	
Fe-59		2,17E+05	
Co-60	**2,5E-03**	**1,26E+07**	
Zn-65			
Sr-89			
Sr-90	**5,2E-05**	**0**	
Zr-95			
Nb-95			
Ag-110m			
Sb-122		4,72E+05	
Sb-124		1,41E+05	
Sb-125			
Cs-134			
Cs-137	**1,4E-03**	**0**	
Ba-140			
La-140			
Ce-141			
Ce-144			
Pu-238			
Pu-239+Pu-240			**MDAs are not reported by NPP**
Am-241	**5,0E-06**	**0**	
Cm-242			
Cm-243			
Cm-244			
Total alpha			
Iodines			
I-131	**2,0E-03**	**1,49E+06**	
I-132			
I-133		1,06E+06	
I-135			
Tritium	**8,0E-01**	**4,27E+11**	
Carbon-14		8,76E+11	

Table IV

Liquid discharges from nuclear power reactor	
Reactor site (name/type): Olkiluoto NPP, BWR	**Period (year of discharge):** 2008
Water volume released during the period (m^3): 2,8E+04 m^3	

Category/Radionuclide	**Highest value of detection limit actually achieved for key nuclides (Bq/m^3)**	**Activity discharged per year (Bq)**	**Commentary**
Other nuclides (excluding H-3)			
S-35			**MDAs are not reported by NPP**
Cr-51		2,26e+07	
Mn-54		6,51e+07	
Fe-55			
Co-58		3,20e+07	
Fe-59		1,01e+07	
Co-60	**1,3E+03**	**1,49e+08**	
Ni-63			
Zn-65			
Sr-89			
Sr-90	**2,0E+02**	**0**	
Zr-95		1,79e+06	
Nb-95		2,33e+06	
Ru-103			
Ru-106			
Ag-110m			
Sn-113			
Sb-122		1,49e+06	
Te-123m			
Sb-124		9,8e+06	
Sb-125		1,03e+07	
I-131			
Cs-134		6,58e+05	
Cs-137	**4,0E+02**	**3,61e+07**	
Ba-140			
La-140			
Ce-141			
Ce-144			
Pu-238			
Pu-239+Pu-240			**MDAs are not reported by NPP**
Am-241	**1,0E+00**	**0**	
Cm-242			
Cm-243			
Cm-244			
Total alpha			
Tritium		2,39E+12	

Annex 4

Information from Sizewell B (U.K.) Regarding Discharge Measurements

Discharges to the atmosphere

All discharges to the atmosphere are assessed by measurement of samples that are collected continuously from the discharge stack. Noble Gas activity is assessed by on-line measurement techniques, while all other radionuclide groups are assessed by taking the samples collected to the on-site Radiochemistry Laboratory for measurement.

Radionuclide group: Noble Gas

The following is reproduced from the certificate of authorisation to discharge radioactivity from Sizewell B Power Station:

For the purposes of demonstrating compliance with the limitations and conditions of this Authorisation relating to "noble gases", the Operator shall measure the activity of noble gases in disposals of radioactive gaseous waste by:

- For the major outlets "Unit Vent " and "Radioactive Waste Building Stack", a Merlin-Gerin Provence Instruments Particulate-Iodine-Gas (PIG) Monitor, drawing a sample from the discharge outlet for analysis using an NGM 20-21 ionisation chamber monitor in which:

i. the detector is set to measure beta radiation in the energy range 250 keV – 3 MeV;

ii. the detector output is routed to a measuring unit, type CM/CI, which passes voltage pulses to the ratemeter, INR, within which the sample specific activity is calculated according to the manufacturer's algorithm;

iii. the system is set up in accordance with the manufacturer's requirements;

iv. the counter is calibrated for detection efficiency using a caesium-137 source traceable to National Standards;

v. the detection range is 1.0 kBq m^{-3} to 1.0 GBq m^{-3} in the beta particle energy range of 250 keV to 3 MeV; and

vi. the measured specific activity of the gas is multiplied by the volume of gas discharged during the reporting period, as measured by a calibrated flow meter;

and

- For the Major Outlet "Gaseous Radioactive Waste System Stack" (GRWS), a Merlin-Gerin Provence Instruments INGM02 Monitor, drawing a sample from the discharge outlet for analysis by an INGM02 Scintillation/sodium iodide detector in which:

i. the detector is set to measure gamma ray emissions in the range 200 keV to 2.5 MeV;

ii. detector output is routed to a measuring unit, type CM/PM, which passes voltage pulses to the process computer where the algorithm calculates the total activity per cubic metre of

discharged gas for the radionuclides krypton-85, krypton-88, xenon-133m, xenon-135 and xenon-135m;

iii. the system is set up in accordance with the manufacturer's requirements;

iv. the gamma spectrometry system is calibrated using caesium-137, and cobalt-60 sources traceable to National Standards;

v. the detection range is 0.4 MBq m-3 to 40 GBq m-3;

vi. the result is corrected for the presence of other radionuclides by comparison with the maypack filter used in the assessment of halogens. Isotopes of xenon or krypton detected on the maypack that cannot be detected by the INGM02 are assessed against the authorisation pro rata, by comparison with those that are detected both on the maypack and by the INGM02; (vii) the derived specific activity of the gas is multiplied by the volume of gas discharged during the reporting period, as measured by a calibrated flow meter;"

The designs of the instruments and the flow rate through the systems being sampled dictate that the Limits of Detection (LoD) are:

- the LoD is 1 kBq m^{-3} at a stack flow rate of 42 m^3s^{-1} for the Radwaste Building HVAC Stack and 50 m^3s^{-1} for the Unit Vent Stack.

- The LoD for the GRWS is 0.4 MBq m^{-3} for a stack flow rate of 1 m^3 $hour^{-1}$.

Radionuclide group: Beta-emitting radionuclides associated with particulate matter

The following text is reproduced from the site's Techniques Document and includes the wording of the discharge authorisation:

Particulate discharges are sampled by drawing an isokinetic sample (the velocity of the gas into the sample head is equal to the velocity of the discharging gas) from a discharge stack through a 5.5 cm diameter glass fibre filter paper, which has a 98% sampling efficiency for particles of 1.6 micron (µm). For the Main Unit Vent and Radwaste Building HVAC Stack systems, the filter paper is inserted into the flow of discharging gas by the use of sampling probes. For the GRWS system, a small portion of discharging gas is taken down a gently curved sample pipe to a filter paper in a purpose-designed holder. In both cases the sampling equipment is carefully designed so as to minimise any disturbance to the flow at the point of sampling, which could otherwise affect the transfer of particulate onto the filter paper. To prevent the filter papers tearing, the sampling equipment is carefully designed to retain and support them in the gas flow.

The gross beta activity of all particulate samples collected by the particulate sampling probes on the Main Unit Vent, Radwaste Building HVAC stack and GRWS stack is analysed off-line in the on-site Radiochemical Laboratory. All samples are subjected to an initial count and if elevated activity is detected, samples are also counted by gamma spectrometry to determine whether the enhanced activity is due to artificial or naturally occurring radioactive materials, the latter being primarily radon daughter products. All samples are then re-counted after 72 hours to allow for the decay of radon daughters and these results are used for the statutory discharge records.

Samples are counted by a scintillation detector based counting system which:

i. has a scintillation detector with a window of maximum total thickness of 1.2 mg/cm2, which is shielded from background radiation using a lead castle;

ii. uses the top shelf position of the lead castle (i.e., the shelf closest to the detector) to count samples;

iii. has a scaler/counter of a type appropriate to be used with a scintillation detector;

iv. is set, using a chlorine-36 source, to have an operating voltage complying with manufacturer's instructions (normally at the mid-point of the plateau);

v. has the lower energy threshold control set at the minimum consistent with appropriately limiting the counting of electronic noise, and the upper energy window control set at maximum so as to have no upper limit set; and

vi. is calibrated for detection efficiency using a chlorine-36 standard traceable to a National Standard.

The LoD is determined by counting statistics. Where activity is not detectable above the background level, a Minimum Detectable Activity (MDA) of 4.65 Standard Deviations of the background measurement is assessed. This typically gives assessed discharges at the LoD of tens of Becquerels per week from the GRWS and tens of kilo-Becquerels per week from the Radwaste HVAC Stack and Unit Vent Stack.

Radionuclide group: Iodine-131

The isokinetic samples drawn from the stacks for the assessment of beta-emitting radionuclides associated with particulate matter are also passed through filter casings filled with approximately 45 grams of activated charcoal, known as a "Maypack" filter. The filters are removed to the on-site Radiochemistry Laboratory once per week, where they are counted by gamma spectrometry using High Purity Germanium Detectors (HPGe). The quantity of iodine-131 detected is decay corrected to the mid-point of the sampling period. Where no iodine-131 is detected, the MDA conservatively is used to estimate the activity discharged. The amount of iodine-131 discharged during the sampling period is assessed pro rata from the ratio of the discharged volume to the sampled volume of gas.

The MDA for a one-thousand second count period is approximately 0.7 Bq of iodine-131, giving a LoD of less than 0.1 MBq per week from the Unit Vent and Radwaste Building HVAC stacks and a LoD of tens of Becquerels from the GRWS stack.

Radionuclide group: Tritium

Samples are drawn from all three stacks continuously for a one week sample period. The sample is fed to an oxidation furnace where tritium (H-3) and carbon-14 (C-14) compounds are oxidised to tritiated water vapour and carbon dioxide respectively. The Unit Vent and Radwaste Building HVAC systems contain enough air to allow the furnace to operate without the need for additional air for combustion, and the furnaces operate at approximately 1000° Celsius. However, the GRWS is operated to exclude air as it is a hydrogen-containing system, so clean air has to be added to the furnace to allow the chemical "oxidation" reaction to take place. The furnace on the GRWS sampling system operates at approximately 400°C, with combustion aided by a catalyst.

Each system has a bubbler train consisting of three gas washing bottles in series. The first two bottles contain 400 millilitres (ml) of 0.1M nitric acid (for tritiated water vapour exchange). Acid is used as a sampling medium to inhibit the removal of carbon dioxide from the gas at this point, as the carbon dioxide will be used subsequently for carbon-14 analysis. The nitric acid is made using water that has been first treated to remove any dissolved chemicals, ensuring a high purity and maximising its ability to absorb the tritium. Upon leaving the first bottle, the residual gas is passed through a second bottle to collect any evaporative losses from the first bubbler and to ensure that nearly 100% of the tritium is removed from the sample gas.

Over the course of a one-week sampling period evaporation occurs so that, after a few days, the first (tritium) bubbler is replaced with a fresh, full bottle. At the end of the sampling period the contents of the three tritium bubbler bottles (two primary bottles and the one secondary bottle) are bulked. Experimentation has shown that 98% of the tritium is retained in these three bottles, with only 2% being carried over into the carbon-14 bubbler bottle (tritium losses from this last bottle being low). The tritium loss is accounted for in the subsequent calculation of the discharge. When samples are changed, fresh nitric acid and clean bottles are used for each sample.

The bulk sample is distilled and the distillate analysed using liquid scintillation. The liquid scintillation counter analyses the samples using an energy channel of 0 to 18.6 keV and verified quench curves to relate the tritium efficiency to the transformed spectral index of the sample. The sample rack is loaded with the blank, a calibrated standard, the samples and a second calibrated standard. The standards are compared with their expected values and the blank is checked to ensure the counter is not experiencing a high background. The counter prints out results in Bq ml^{-1} of distillate. Having allowed for the sampling losses, the total amount of tritium collected in the sample over the sampling period is converted into tritium activity discharged from the ratio of the discharged volume to the sampled volume of gas.

The MDA is 0.016 Bq ml^{-1} of distillate, corresponding to a LoD for tritium in gases of tens of kilo-Becquerels from the GRWS stack and tens of mega-Becquerels from the Unit Vent and Radwaste Building HVAC stacks.

Radionuclide group: Carbon-14

The tritium sampling systems described above are also used for the sampling of carbon-14. After the sample gas has passed through the second nitric acid bottle, it is passed into a third glass bottle containing 300 ml of 1M sodium hydroxide solution for CO_2 removal. Passing the sample gas through the furnace and mixing it with air at a high temperature ensures that carbon-14 is fully oxidised, which then ensures that it reacts chemically with the sodium hydroxide solution using the reaction:

$$CO_2 + 2NaOH \rightarrow Na_2CO_3 + H_2O$$

The sodium hydroxide solution is made using demineralised water. Although the carbon-14 bottle is also prone to evaporation, this merely has the effect of concentrating the caustic and does not affect its ability to trap carbon dioxide. Neither does the evaporation process release carbon that is already trapped, as it is chemically bound in the form of sodium carbonate. The fraction of any carbon dioxide undergoing the above reaction, compared with the total amount of CO_2 in the sample, is called the Trapping Efficiency.

A 100 ml aliquot is taken from the carbon-14 bottle, and precipitated with approximately 25 ml of barium chloride to form insoluble barium carbonate using a No. 42 Whatman filter or No. 4 glass crucible, using the reaction:

$$Na_2CO_3 + BaCl_2 \rightarrow BaCO_3 + 2NaCl$$

The total amount of precipitate is washed with water and methanol, dried and weighed. One gram of the precipitate is taken into suspension with 8 ml of demineralised water and 8 ml of the liquid scintillation cocktail and counted against an appropriate carbon-14 standard. Two such samples are prepared in glass vials. A "blank" is prepared using non-radioactive barium carbonate in suspension. Calibrated standards are prepared from standard solutions of $Na_2{}^{14}CO_3$.

The liquid scintillation counter analyses the samples using the energy channels of 0 to 18.6 keV and 18.6 keV to 154 keV, with the counts in the second channel being used for the calculation. The sample rack is loaded with the blank, a calibrated standard, the samples and a second calibrated standard. The counter prints out results in counts per minute (cpm) for each sample. The majority of the carbon dioxide passing through the bubbler is not radioactive and can be assumed to be at the concentration typical of clean air (0.033% by volume). The activity in the aliquot is corrected for the Trapping Efficiency and sampling efficiency of the system, by calculating how much precipitate would have been formed under ideal (100% efficient) conditions. This is used as a correction factor, along with the stack flow to sample flow ratio, to calculate the activity discharged.

LoD for the measurements are typically < 10 kBq for the GRWS stack and a few MBq for the Radwaste Building HVAC and Unit Vent Stacks.

Discharges to the Sea

All disposals to sea are routed to the surge chamber, an open pit into which the cooling water from the turbines is channelled, providing dilution at up to 58 m^3 s^{-1}. Disposals from the turbine hall ("secondary liquid discharges") are treated as though they may be radioactive and are subject to the same analyses for radioactivity as are the disposals from the Radioactive Waste Building ("primary side discharges").

Waste is collected in a series of tanks, sampled and analysed, then processed according to the findings of the analysis. The waste is forwarded to final monitoring tanks for a final sample and analysis for the purpose of authorising the discharge. The sample used for reporting to the regulator and for assessment against the authorised discharge rates is taken during discharge using a flow proportional sampler. This sampler is equipped with a shuttle valve that takes repeated aliquots from the discharge pipe line as each discharge is carried out, with the total volume sampled being proportional to the volume discharged.

All of the samples from primary side discharges taken over the course of a week are aggregated, and all of the samples from the secondary side are aggregated separately. Aliquots of the aggregated samples are processed for analysis as described in the following sections.

An additional aliquot of the primary side aggregate over three months is sent to an off-site laboratory for more detailed isotopic analysis, primarily by gamma spectrometry. An annual aggregate of these samples undergoes further analysis to include non-gamma emitting radionuclides such as iron-55 and nickel-63.

Radionuclide group: Tritium

An aliquot of the sample is distilled and the distillate collected for analysis. If the distillate is likely to have a tritium specific activity greater than 5,000 Bq/ml, as determined by the analysis of pre-discharge samples during the sampling period, an aliquot of the distillate is diluted with ultra-pure

water to ensure that the liquid scintillation counter detectors are not saturated, which would cause under-reading of the sample. The amount of dilution is increased if the sample is likely to be greater than 50,000 Bq/ml.

Four ml of the prepared distillate is taken into each of two 20 ml vials with 12 ml of scintillant.

The samples are counted by liquid scintillation, using energy channels of 0 to 18.6 keV, 18.6 to 167 keV and 167 to 2000 keV. Only a tritium standard is used, the external standard quench correction method being used. The tritium count rate is taken from the low energy channel.

The output from the instrument in disintegrations per minute (dpm) is averaged for the two vials and corrected for dilution, if any. This value is multiplied by 60 seconds and divided by 4 ml to give a result in Bq/ml, equivalent to MBq/m^3. This value is multiplied by the total volume of effluent discharged during the sampling period to determine the activity discharged.

The LoD value is typically 0.016 Bq/ml.

Radionuclide group: Caesium-137

A 20 ml aliquot of the sample is taken into a plastic bottle of the calibrated geometry and counted directly using gamma spectrometry. The activity of caesium-137, or its LoD value, is divided by 20 ml to get a result in Bq/ml, equivalent to MBq/m^3. This value is multiplied by the total volume of effluent discharged during the sampling period to determine the activity discharged.

The LoD value is typically 0.02 Bq/ml for the primary discharges. The secondary side discharge samples are counted for a longer period to reduce the LoD, and the LoD is typically 0.005 Bq/ml. This is because the larger volumes of waste discharged from the secondary side would dominate caesium-137 discharges, if the higher LoD were to be used, as the LoD is used as the MDA to conservatively calculate the activity discharged.

Radionuclide group: Other Radionuclides

Note that this is another radionuclide group for which the value reported depends upon the technique used for measurement. As such the details of the technique are given in the discharge authorisation.

Five ml of the sample is taken into each of two fresh glass liquid scintillation counter (LSC) vials. One ml of 6 M nitric acid and a few drops of 30% (w/v) hydrogen peroxide are added.

The solution is evaporated in the glass vials under an infrared lamp until the vials are completely dry. The evaporate is re-dissolved in 10 ml of ultra-pure water and the evaporation process repeated, to ensure complete removal of tritium. The second evaporation has been shown to be necessary at high tritium specific activity and/or high sample boron concentration. Complete removal of tritium from the sample is necessary to prevent the presence of low energy beta-emitting and electron-capture nuclides from being masked.

Following tritium removal, the evaporate is re-dissolved in 4 ml of 0.1M nitric acid and 12 ml of scintillant is added.

The vial is counted using a triple channel LSC technique, the counter having been set and calibrated as below:

- the “low energy” channel of the liquid scintillation spectrometer set to detect all beta energies below 8.0 keV, calibrated for detection efficiency using a tritium standard;

- the “intermediate energy” channel set to detect all beta energies above 18.6 keV and up to the maximum energy of 167 keV, calibrated for detection efficiency using a carbon-14 standard;

- the “high energy” channel set to detect all beta energies above 167 keV and up to the maximum detectable by the counter (2000 keV), calibrated for detection efficiency using a standard consisting of strontium-90 and yttrium-90;

- the “automatic energy/quench correction” function applied to the counting protocol.

The output from each energy region, in dpm is multiplied by 60 seconds and divided by 5 ml, to give the sample specific activity in Bq/ml, equivalent to MBq/m^3. This value is the “total activity excluding tritium”. The specific activity of caesium-137 measured earlier is then subtracted to give the specific activity of Other Radionuclides in the sample.

This value is multiplied by the total volume of effluent discharged during the sampling period to determine the activity discharged.

The LoD is 0.01 Bq/ml for both primary side and secondary side discharges.

Annex 5

United Kingdom and United States Studies of Historic Nuclear Reactor Discharge Data

United Kingdom

The results of a study of discharge rates from selected currently operating power plants of four reactor types were published by D. Copplestone et al. (Study of historic nuclear reactor discharge data, Radioprotection, volume 44, no. 5, 2009, pages 875-880). The objective was to assess BAT for discharges for a potential new-build reactor using discharge data from operating plants of similar reactor type. The study concluded that "there is no simple, clear, or easily explained relationship between radioactive discharges and reactor power output". In some cases, evidence seemed to suggest proportionality between discharges and power output, but that was not true for all cases examined. The report acknowledges that the magnitude of discharge may be affected "as a result of an abnormal event", and efforts were made to identify any such events.

Table 2 of the report by D. Copplestone et al. is shown below. The data is based on discharge rates from at least six operating units considered to be predecessor units for proposed new (and improved) reactor designs. Where possible, data were collated for an operational period of ten years. The "maximum GBq per GWeh" column reflects the value of the mean plus one standard deviation. That value "was then used to approximate the range of radioactive discharges that might be expected during normal operations".

Table 2 Mean and standard deviation of the data available for the predecessor designs

Design	Waste Stream	Mean GBq/GWeh	Standard GBq/GWeh deviation	Maximum GBq/GWeh	Predicted GBq/GWeh
AP1000	Liquid tritium	3.03E+00	1.58E+00	4.61E+00	3.82E+00
	Other liquids	2.18E-03	2.44-03	4.62E-03	9.69E-04
Total liquid		*3.03E+00*	*1.58E+00*	*4.61E+00*	*3.82E+00*
	Airborne tritium	2.12E-01	3.26E-01	5.38E-01	1.32E+00
	Airborne noble gases	2.80E-01	4.16E-01	6.96E-01	4.17E+01
	Airborne iodine-131	1.35E-05	4.34E-05	5.69E-05	4.54E-04
	Airborne particulates	2.72E-06	5.45E-06	8.17E-06	1.79E-04
	Airborne carbon-14	1.80E-02	8.60E-03	2.66E-02	2.76E-02
Total airborne		*5.10E-01*	*1.11E+00*	*1.62E+00*	*4.30E+01*
EPR	Liquid tritium	1.68E+00	7.50E-01	2.43E+00	3.58E+00
	Other liquids	6.85E-05	1.36E-04	2.04E-04	1.62E-03
Total liquid		1.68E+00	7.50E-01	2.43E+00	3.58E+00
	Airborne tritium	8.50E-02	5.90E-02	1.44E-01	3.42E-02
	Airborne noble gases	4.81E-01	7.19E-01	1.20E+00	5.50E-02
	Airborne iodine-131	1.05E-06	1.95E-06	3.00E-06	1.57E-06
	Airborne particulates	1.48E-07	2.32E-07	3.80E-07	2.75E-07
	Airborne carbon-14	3.07E-02	1.39E-02	4.46E-02	2.41E-02
Total airborne		5.97E-01	7.93E-01	1.39E+00	1.13E-01
ESBWR	Liquid tritium	4.04E-02	3.25E-02	7.29E-02	3.79E-02
	Other liquids	1.37E-04	1.88E-04	3.25E-04	2.65E-04
Total liquid		*4.05E-02*[1]	3.27E-02[1]	*7.32E-02*	*3.82E-02*
	Airborne tritium	6.53E-02	8.37E-02	1.49E-01	2.05E-01
	Airborne noble gases	3.92E-01	6.18E-01	1.01E+00	1.12E+01
	Airborne iodine-131	2.82E-06	5.24E-06	8.06E-06	1.10E-03
	Airborne particulates	3.42E-05	1.52E-04	1.86E-04	3.59E-04

	Airborne carbon-14	Not available	Not available	Not available	Not available
Total airborne		*4.57E-01*[1]	7.03E-01[1]	*1.16E+00*	*1.14E+01*
ACR1000	Liquid tritium	3.74E+01	3.73E+01	7.47E+01	1.26E+01
	Other liquids	2.03E-03	2.37E-03	4.40E-03	1.47E-03
Total liquid		*3.74E+01*[2]	3.73E+01[2]	*7.47E+01*	*1.26E+01*
	Airborne tritium	3.68E+01	4.12E+01	7.80E+01	5.26E+00
	Airborne noble gases	1.44E+01	4.24E+01	5.68E+01	1.68E+00
	Airborne iodine-131	5.66E-06	1.10E-05	1.67E-05	8.42E-07
	Airborne particulates	1.24E-05	5.35E-05	6.59E-05	Not available
	Airborne carbon-14	1.81E-01	2.36E-01	4.17E-01	2.95E-02
Total airborne		*5.14E+01*[2]	*8.36E+01*[2]	*1.35E+02*	*6.97E+00*

As part of the development of this report for the Expert Group on Best Available Techniques, the mean values for discharge rate per GWe-year were entered in a screening tool to estimate doses to members of the public. (The work was performed under the direction of one of the report's co-authors, A. Sutherland.) Basic assumptions included the following: Noble gases were taken to be krypton-85, airborne particulate was taken to be caesium-137 with an AMAD of one micron, release to the atmosphere was via a 30-metre-tall stack, other liquids was taken be caesium-137, and a receiving river flow of 50 cubic metres per second was used.

The estimated annual dose from any of the historic normalised mean discharge rates was well below the dose constraint of 0.3 mSv annual dose to a member of the public that is used in the United Kingdom. Releases to the atmosphere of noble gases, iodine-131, and particulate matter each resulted in a calculated annual dose of less than 0.1 micro-Sv. The table below lists other calculated doses related to releases to the atmosphere.

Table A5-1 Calculated Doses due to Predecessor-Unit Releases to the Atmosphere using the United Kingdom Screening Tool

Proposed New Design	Radionuclide	Calculated Dose from Predecessor Units (microSieverts/a)
AP 1000	Tritium	0.1
	Carbon-14	1.1
EPR	Tritium	0.05
	Carbon-14	1.8
ESBWR	Tritium	0.04
	Carbon-14	Not Available
ACR 1000	Tritium	21
	Carbon-14	11

As can be seen, each calculated annual dose (estimated to be resultant from releases to the atmosphere) for each of the reactor types except the ACR 1000 is less than 1% of the dose constraint

used in the United Kingdom for a member of the public. For the ACR 1000, each calculated annual dose from the predecessor units is less than 10% of the dose constraint used in the United Kingdom.

Regarding releases to aquatic systems, the table below lists the calculated annual doses.

Table A5-2 Calculated Doses due to Predecessor-Unit Releases to Water using the United Kingdom Screening Tool

Proposed New Design	Radionuclide	Calculated Dose from Predecessor Units (microSieverts/a)
AP 1000	Tritium	0.3
	Other (Cs-137)	3
EPR	Tritium	0.2
	Other (Cs-137)	0.2
ESBWR	Tritium	0.04
	Other (Cs-137)	0.4
ACR 1000	Tritium	4
	Other (Cs-137)	5.8

For releases to water, each calculated annual dose is less than 2% of the dose constraint used in the United Kingdom.

United States

The results of a study of discharge rates from commercial nuclear power plants operating in the United States between 1995 and 2005 were published by J. Harris and D. Miller (Radiological Effluents Released by U.S. Commercial Nuclear Power Plants from 1995-2005, Health Physics, vol. 95, no. 6, 2008, pages 734-743). The objective of the study was to compile the U.S. industry data, identify trends, and calculate average population dose commitments for that timeframe. Discharge data were compiled in the categories recognized by the U.S. Nuclear Regulatory Commission for annual data reporting. Because discharge rates to aquatic systems were small for fission and activation products and dissolved and entrained gases, those two categories were summed and listed as "other radionuclides" when analyzing liquid effluents.

A result of the study was that for "nearly every category for both reactor types [PWR and BWR, the trend] is fairly level in terms of activity released for the entire time period. One exception was found for PWR particulate releases to the atmosphere, for which one event in 2003 skewed the industry data, "especially since the annual radioactivity released in particulate matter is so low compared to tritium or fission and activation gases". The following year saw a significant drop in the entire industry radioactivity followed by another sharp increase. This increase in 2005 was due to higher activity level releases by several plants". As for liquid effluent, "liquid releases have stayed very constant over the 11-y period". The authors went on to state that "improvements in radioactive

waste treatment and reactor operations are offset by increased power production, increased capacity factors, and power up-rates".

Eighteen release categories were evaluated for trends over the analyzed time period. For 12 of the 18 categories, no significant trend over time was observed in terms of activity released. For 3 of the 18 categories, a trend toward reductions in release rates was found, and for the other 3 categories, a trend toward increases in release rates was found. In support of the statement at the end of the previous paragraph, it was tabulated that electrical energy produced (GW) over the period from 1995-2005 generally increased from figures in the range of 75 (1995-1998) to 88 (2002-2005). As for calculated doses to members of the public, the authors observed that "the study showed how low normal operation effluent doses are, compared to regulatory limits".

Annex 6

Examples of Potential Abatement Techniques

Abatement techniques may be used in combination to achieve optimum reductions in release rates. Some listed technologies may be relatively mature and others may be considered to be emergent and/or not necessarily available for near-term use on an industrial scale. Brief, informal description of each technology is listed; such information should not be considered definitive for any use generally or in a specific, local circumstance. Effective implementation of selected technologies (e.g., via appropriate design, construction, operation, and maintenance) is needed for optimal use of those technologies.

Abatement Technologies for Aqueous Solutions

Biotreatment – use of biological materials to accumulate contaminants or alter their physico-chemical form for alternative treatment or release

Centrifuging – removal of solid material from solution by rapidly rotating a solution in a vortex and causing particles to migrate to the outside wall of a hydrocyclone

Chemical precipitation – removal of radionuclides dissolved in solution by adjustment of pH or use of chemical reactions

Cobalt-60 reduction – use of design changes such as revised material composition and/or feedwater iron control to reduce the production of cobalt-60 by activation of cobalt-59

Evaporation – use of elevated or ambient temperature evaporation to reduce volumes of effluent and collect residue for alternative treatment or release

Filtration – use of filtering media to remove insoluble materials suspended in solutions; removal may be designed for larger particles down to colloidal materials

Ion exchange – use of specialty resins or other media to remove lower levels of specific contaminants from solutions; often used after mechanical filtration of the solution

Passivation – use of deposited materials (such as depleted zinc oxide) in part to reduce general corrosion rates of primary system materials and/or improve ability to remove radionuclides via the selected combination of abatement technologies

Reverse osmosis – use of permeable membranes to remove lower levels of small particulate contaminants from solutions

Abatement Technologies for Airborne Material

Adsorption – use of media such as carbon filter beds with high effective surface areas to remove volatile chemically reactive gases from the air flow

Cryogenics – use of low temperatures to collect selected contaminants for alternative treatment or release

Filtration – use of filtration media (such as HEPA filters) to remove particles suspended in the air flow

Hold-up or Delayed release – use of piping systems or tanks to delay the release of gaseous effluent, thereby allowing for the radioactive decay of shorter-lived radionuclides in-plant

Scrubbing – use of wet gas scrubbing by materials such as sodium hydroxide solution to remove particulate matter from the air flow

Abatement Technologies for Solid Materials

Hold-up – use of tanks and/or storage areas to delay the release or shipment offsite of solids or solutions with high solid content, thereby allowing for the radioactive decay of shorter-lived radionuclides on the plant site

Incineration – use of thermal treatment of waste, to reduce the volume of solid waste to be shipped to a storage and/or disposal facility

Transformation, transmutation – use of technologies to change a radionuclide or its chemical form into another nuclide or chemical form which would be of lesser potential impact on the environment if/when released

Annex 7

Ancillary Strategic Issues to the Implementation of BAT

Two principal areas may be categorised as strategic issues, in the sense of their scale and broad nature, which may initially be perceived as outside the scope of the implementation of BAT. Nevertheless, they are in fact likely to be issues for achieving and maintaining implementation of BAT.

First of these is loss of expertise or knowledge from the industry: the best technology and good management are unlikely to deliver good results if there is a lack of relevant specialist expertise amongst staff of operators and regulators. Secondly, while historically nuclear power plants have often been operated under the auspices of the government (and so have been centrally controlled), developments will be more and more shaped by the market place. This second issue is linked to the first, notably in that exposure of nuclear operators to market forces has resulted in an increasing use of contractors, which raises issues for loss of expertise within operating companies and the need for sharing of potentially scarce resources across operating companies.

Loss of expertise

To effectively manage safety and environmental aspects of a nuclear power plant, a broad range of expertise is needed and, considering a plant from initial design through decommissioning also highlights the need for suitably qualified and experienced personnel away from the plant itself, for example, in regulatory and/or waste management organisations.

Presently, three processes are taking place that increase the demand for people with suitable qualifications and experience:

- decommissioning or planning for decommissioning of nuclear facilities no longer to be operated;
- foreseen building of new nuclear power plants;
- extended operating lifetimes for existent nuclear power plants.

Declining interest in nuclear energy in the 1980s and 1990s, together with the demographics (e.g., retirements of personnel in operating companies and regulatory agencies) has compounded this potential large increase in demand for personnel with a reduction in the pool of people with suitable qualifications and experience.

On the other hand the foreseen construction of nuclear power plants with an expected operating lifetime of 60 years or more poses challenges to maintaining satisfactory expertise during this long period, as well as expertise required before and after the period of operation. A system of maintaining the expertise among the staffs of new nuclear power plants should take into account the fact that several generations of workers would actually work at the same site. Relevant considerations here might include use of apprenticeships, level of employee retention and record keeping (e.g., what is kept, for how long, and will the storage media be readable throughout the required time period).

Impact of the market place

The development of a free market and deregulation in the nuclear field has taken place in many countries. The consequences of a free market may have very positive effects, as for example lowering the cost of production of electricity, but can also have negative effects. These negative effects, which could influence the application of BAT as well as safety more generally, is a challenging issue for regulatory authorities in countries where deregulation is quite new. These authorities should systematically recognise the influences before they appear and prevent a decline in standards. Three issues related to development of a market place potentially could lead to a decline of the safety and effective operation:

- larger involvement of contractors and outsourcing;
- decentralisation of the nuclear fuel cycle;
- ownership with no background related to operation of nuclear power plants.

Larger involvement of contractors and outsourcing

In many countries contractors are very much involved in the operation of a nuclear power plant not only during its modernisation but also during maintenance or a refuelling period. In some nuclear power plants there is even a trend to increasingly involve contractors in everyday activities. The reason for this greater involvement is economic. The work done by contractors ranges from specialised work, as for example testing reactor coolant pumps, to relatively less specialized jobs, such as painting in controlled areas.

The results of involvement of contractors could have influence on the basic performance of a nuclear power plant. The operator as well as the regulator should recognise this influence. The mutual cooperation between contractor and operators is needed to control particular aspects of a nuclear power plant's safety and environmental performance.

By the nature of their work scope, contractors tend to be concerned with one aspect of running a nuclear power plant whereas the operator has to maintain a broader view, concerning all phases of a plant's life, including planning for decommissioning. The influence of contractors on waste generation as an important element in implementing BAT as related to discharge abatement should be carefully studied and controlled. The environmental safety culture of a contractor can have an influence on the overall environmental safety culture of plant personnel: if a contractor fully supports and applies the plant operator's environmental safety culture, they may be a positive influence on overall performance of the facility.

Use of contractors and outsourcing also relates to knowledge management, since if specialist contractors move from job to job on an *ad hoc* basis, the operator may in effect have a gap in their knowledge. Contrastingly, contractors may have limited familiarity with a particular site. A possible strategy to handle this is to ensure that specialist contractors are engaged in long-term partnerships with the operator.

Decentralisation of the nuclear fuel cycle

In the past, nuclear fuel cycle facilities were very often under central ownership or control, usually of a government or a government-owned company. In many countries this is no longer the case, and decentralisation has taken place. Many different organisations have become owners of

nuclear facilities. Decentralisation can potentially lead to gaps or discontinuities in control over some parts of the nuclear cycle. The owners of the various facilities and the regulatory authority or authorities should recognise the potential for isolation of goals and programmes as decentralisation occurs, and the consequent potential influence on safety, quality and environmental matters. In particular, separation of ownership could lead to each ownership "segment' being optimised individually, rather than the nuclear fuel cycle being optimised as a whole. Care needs to be taken that safety, quality and environmental aspects are not compromised through any part of the nuclear fuel cycle. This task of ensuring the existence of an integrated, seamless environmental safety programme is even more demanding if in addition to decentralisation of nuclear fuel cycle facility ownership, more than one regulatory authority is involved.

Ownership with limited background related to operation of nuclear power plants

In some countries, nuclear power plants are objects in a free market and can potentially be bought by owners with no experience in the nuclear field. The regulatory authority responsible for radiation and environmental safety should be aware that legislation related to nuclear power plants and implementation of its requirements could be a challenge for owners with limited experience in the field. Change of ownership may go hand-in-hand with management changes, and the regulator should be prepared to assess and possibly act on changes that might adversely affect the application of BAT related to discharge abatement.