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AP1000 Nuclear Power Plant BAT Assessment

UKP-GW-GL-026, Revision 2

Westinghouse Electric Company LLC
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REVISION HISTORY

Revision	Description of Change
0	Initial Submittal
1	Complete rewrite to address Regulatory Observation Actions 34.A02 and 34.A03, and Technical Queries
2	Updated to include relevant information provided in responses to regulator technical queries and regulatory observations from Apr – Dec 2010. Incorporates the following DCP's: • APP-GW-GEE-2083 • APP-GW-GEE-2085

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The UK nuclear regulators have developed a Generic Design Assessment (GDA) process for evaluating alternative designs for the next generation of nuclear power plants to be built in the United Kingdom. Initially the Regulators will review the safety, security and environmental impact of the nuclear power plant designs against a generic site which, as far as possible, envelops or bounds the characteristics of any potential UK site.

The Westinghouse AP1000^{TM} Nuclear Power Plant (AP1000 NPP) AP1000 nuclear design was developed in the U.S. with safety and simplicity and As Low As Reasonably Achievable (ALARA) as the fundamental design principles in contrast to the As Low As Reasonably Practicable (ALARP) and best available techniques (BAT) principles required in the United Kingdom. The standard AP1000 NPP design is well developed, and there is limited scope for further development and design change.

A key issue for the UK environmental regulators is that the AP1000 NPP design has incorporated BAT to minimise the production and discharge of waste. This report identifies the AP1000 NPP design characteristics which contribute to the production, minimisation and treatment of waste and relates these techniques to BAT. The report also presents radionuclide BAT forms that describe the formation mechanisms for key radionuclides and the options for minimisation and abatement of their gaseous and liquid release. It is concluded that the design process has produced a plant that uses BAT techniques for minimising the production and discharge of waste.

The standard AP1000 NPP design is less prescriptive when it comes to the treatment of intermediate radioactive waste (ILW) and low level radioactive waste (LLW). The treatment of ILW and LLW has been subject to optioneering and BAT evaluation and this process is summarized in this report. The assessment concluded that BAT for ILW is the cement encapsulation, long-term storage, and disposal to the national repository. The BAT treatment and disposal of LLW involves decontamination to the maximum extent practicable, segregation of non-radioactive wastes for free release, size reduction and compaction. LLW will be transferred directly into approved waste containers for transfer to the repository.

LIST OF ACRONYMS

ac	Alternating Current
ADS	Automatic Depressurisation System
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
AP1000 NPP	Westinghouse AP1000® Nuclear Power Plant
ASHRAE	American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc.
ASME	American Society of Mechanical Engineers
BAT	Best Available Techniques
BDS	Steam Generator Blowdown System
BEGL	British Energy Generation Ltd.
BPEO	Best Practicable Environmental Option
BPM	Best Practicable Means
CCS	Component Cooling Water System
CDS	Condensate System
CFA	Conditions for Acceptance
CPS	Condensate Polishing System
CVS	Chemical and Volume Control System
CWS	Circulating Water System
DBD	Different By Design
dc	Direct Current
DOP	Di-Octyl Phthalate test (filter systems)
DTS	Demineralised Water Treatment System
EA	Environment Agency
GDA	Generic Design Assessment
HEPA	High Efficiency Particulate Air
HLW	High Level Waste
HSE	Health & Safety Executive
HVAC	Heating, Ventilation and Air Conditioning
ILW	Intermediate Level Waste
IPC	Integrated Pollution Control
IRWST	In-containment Refuelling Water Storage Tank
IVR	In-vessel Retention
LLW	Low Level Waste
LLWR	Low Level Waste Repository
LOCA	Loss-of-coolant Accident
MCR	Main Control Room
MTU	Metric Tonne of Uranium
MWD	Megawatt Day
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
PPE	Personal Protective Equipment
PRA	Probabilistic Risk Assessment
PWR	Pressurised Water Reactor
PWS	Potable Water System
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RWMD	Radioactive Waste Management Directorate
SDS	Sanitary Drainage System
SFAIRP	So Far As Is Reasonably Practicable
SG	Steam Generator
SMPP	Secure Military Power Plant

LIST OF ACRONYMS (cont.)

SWS	Service Water System
TCS	Turbine Building Closed Cooling System
UK	United Kingdom
VAS	Radiologically Controlled Area Ventilation System
VBS	Nuclear Island Non-Radioactive Ventilation System
VCS	Containment Recirculating Cooling System
VDU	Visual Display Unit
VES	Main Control Room Emergency Habitability System
VFS	Containment Air Filtration System
VHS	Health Physics And Hot Machine Shop HVAC System
VRS	Radwaste Building HVAC System
VTS	Turbine Building Ventilation System
VWS	Central Chilled Water System
VXS	Annex/Auxiliary Buildings Non-Radioactive HVAC System
VYS	Hot Water Heating System
VZS	Diesel Generator Building Heating And Ventilation System
WEC	Westinghouse Electric Company
WETOX	Wet Air Oxidation
WGS	Gaseous Radwaste System
WLS	Liquid Radwaste System
WSS	Solid Radwaste System
WWS	Waste Water System

1.0 INTRODUCTION

The UK nuclear regulators (the Health and Safety Executive [HSE] and the Environment Agency [EA]) have developed a generic design assessment (GDA) process for evaluating alternative designs for the next generation of nuclear power plants to be built in the United Kingdom. Westinghouse Electric Company has submitted an application for its AP1000 NPP design to be considered in this process.

The EA has reviewed the preliminary Westinghouse application and commented that the submittal would benefit from a formal assessment of the best available techniques (BAT) for dealing with each significant waste stream released from the AP1000 NPP.

The AP1000 NPP design is not only an evolutionary development of previous pressurized water reactor (PWR) generations, but also incorporates many innovative features that greatly simplify the design and contribute to improvements in safety and performance. This report identifies the AP1000 NPP design characteristics that contribute to the minimisation of the production and discharge or disposal of waste and relates these characteristics to BAT. The standard AP1000 NPP design is less prescriptive when it comes to waste treatment of intermediate level radioactive waste (ILW) and low level radioactive waste (LLW). The treatment of ILW and LLW has been subject to optioneering and BAT evaluation. The methodology is summarised, and the selected BAT option is described.

2.0 OBJECTIVES AND SCOPE

The objective of this report is to collate information from the AP1000 NPP design that describes how the environmental protection concept of BAT is applied to the control of waste emissions and discharges.

The report addresses Reference 1.5 of the EA "Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs" [Reference 1] which is reproduced below:

> An analysis should be provided that includes an evaluation of options considered and shows that the Best Available Techniques will be used to minimise the production and discharge or disposal of waste. This should include:
>
> - a description of the means used by each significant waste generating and management process to minimise waste arising and discharged or disposed of and a demonstration that these are the best practicable;
> - a review of design features, including those of fuel usage, such as burn-up and rating, that facilitate minimisation of arisings and disposal of waste during operation of the reactor;
> - a review of design features that facilitate decommissioning and minimise the arisings of decommissioning waste.
>
> Reference should be made to:
>
> - all periods of "operation", for example at power, shutdown, maintenance and refuelling (including related tasks such as fuel and flask handling);
> - transitory periods (e.g. returning to power following shutdown);
> - issues relating to minimising radioactivity source terms (for example materials of construction and coolant chemistry);
> - abatement issues (for example optimising resin types and usage in treatment systems); process control and monitoring arrangements including fault detection;
> - the selection of materials and physical features to minimise activation and contamination, facilitate decontamination, removal of components etc; and
> - practices at other existing and proposed facilities.

This report also sets out to demonstrate how the principles of the waste hierarchy (see Figure 2-1) and the BAT management factors for the optimisation of releases from nuclear facilities (see Figure 2-2) are implemented to achieve BAT in the AP1000 NPP.

A distinction will be drawn between the BAT aspects of the AP1000 NPP design evolution as applied to AP1000 NPP systems that are now designed and standardised and those systems which are subject to ongoing design (for example, radwaste treatment system).

One of the strengths of the AP1000 NPP is that it offers a single worldwide design to provide advantages in common safety systems and maintenance regimes. This has significant safety benefits with the ability to transfer experience among all AP1000 plants. For this reason, the BAT assessment for the standardized AP1000 NPP systems will be limited in scope to initially describe the history of AP1000 NPP, the alternative techniques that were considered during development and the improvements made to aid the BAT justification (see Section 4). This section also provides an inventory of each significant waste stream and identifies the abatement technologies, process control, monitoring techniques, and describes management techniques used to prevent or minimise releases.

In Section 5, BAT forms have been developed for key radionuclides to demonstrate that the AP1000 NPP design makes comprehensive use of the best techniques for prevention, minimisation, and abatement of radioactive emissions and discharges.

For the radwaste system, where design is in progress, a detailed approach to the BAT assessment has been followed. The BAT approach evaluates indicative BAT and screens out alternative options based on a matrix of technical performance, safety, environmental impact, operability, and cost issues. Favourable options are short-listed, and the more detailed assessment is carried out to determine the final BAT option. This methodology is described in Section 6.

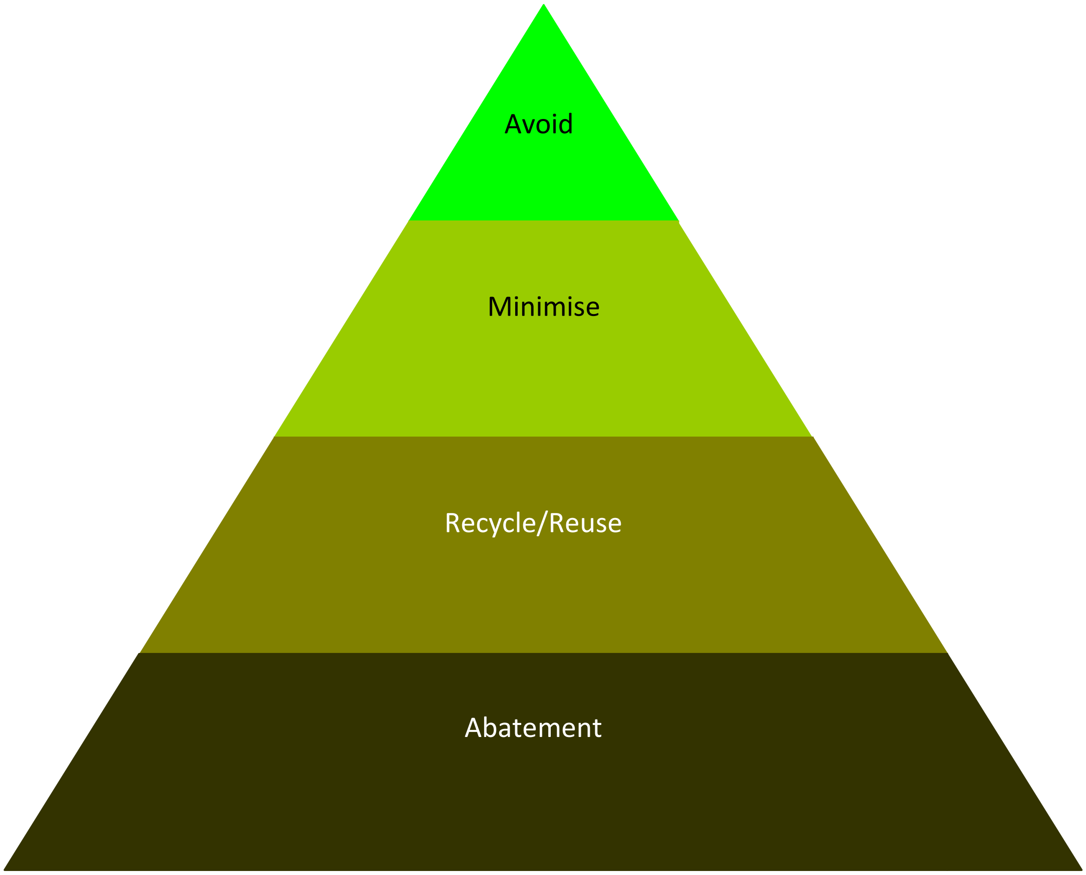


Figure 2-1. Waste Management Hierarchy

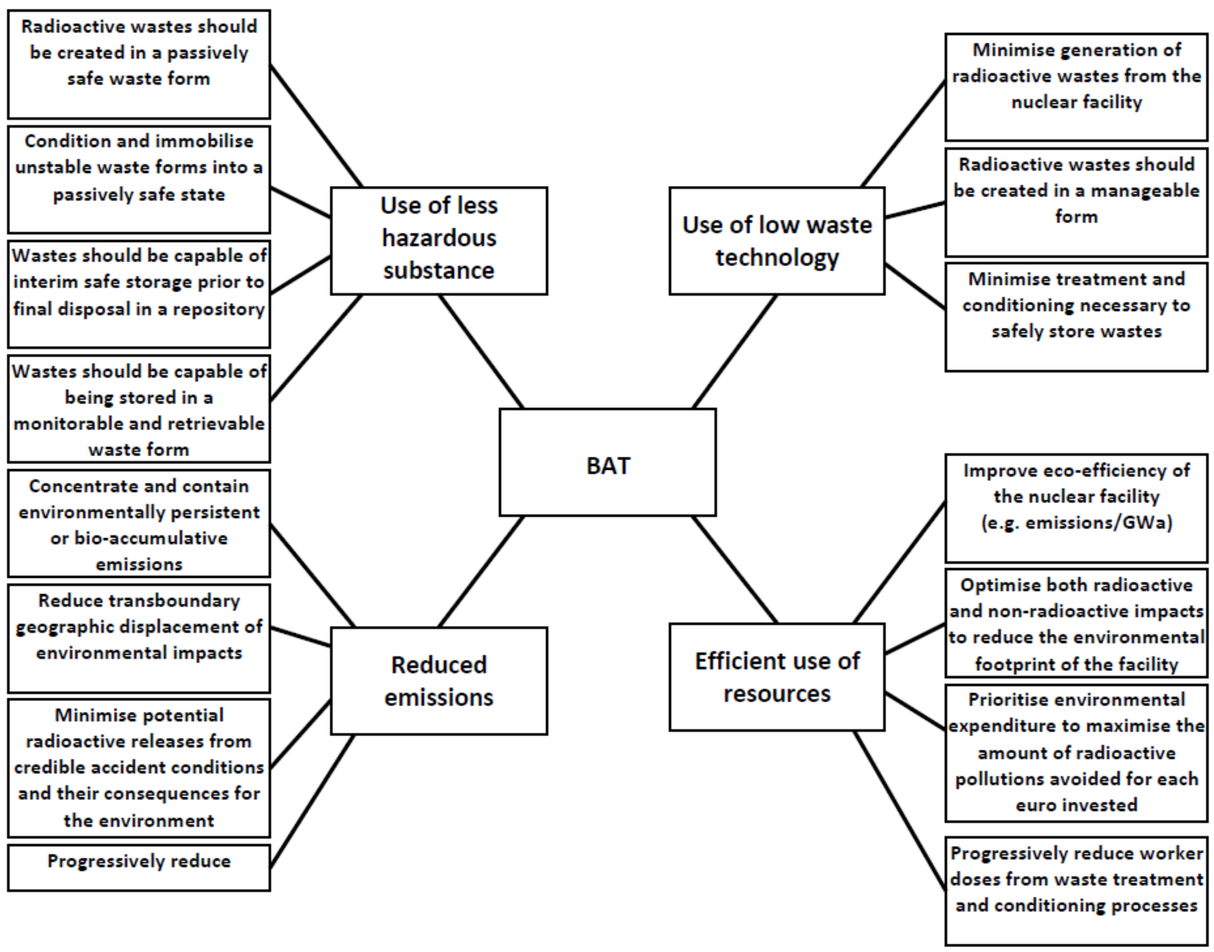


Figure 2-2. Nuclear BAT Management Factors for Optimization of Releases from Nuclear Facilities [Reference 2]

3.0 RELATIONSHIP BETWEEN BAT AND OTHER REGULATORY PRINCIPLES

3.1 Relationship Between BAT, ALARA and ALARP

A cornerstone of the design of the AP1000 NPP is the incorporation of radiation exposure reduction principles to keep worker dose ALARA. This approach is similar to the two principles underpinning UK health and safety legislation that require workplace risks to be reduced to ALARP or so far as is reasonably practicable (SFAIRP). In practice, the HSE consider the terms ALARP and SFAIRP to be interchangeable. The distinction is only made where it is necessary to cite the relevant term used in specific legislation. For assessment purposes the HSE also consider ALARP to be equivalent to ALARA used by other bodies in radiation protection nationally and internationally. The AP1000 NPP design has addressed safety to the public and power plant workers, and the impact on the environment under both normal and accident conditions. The design is fully compatible with ALARA and ALARP.

The difference between ALARP and BAT is primarily one of emphasis: ALARP is driven by health and safety issues, and BAT is a concept associated with environmental protection. In practice, many of the design steps taken in response to ALARP contribute to low environmental emissions, and hence, also relate to BAT. Both BAT and ALARP include the concept of balancing costs against benefits by identifying and selecting processes, operations, and management systems that prevent or minimise releases where it is cost effective to do so. ALARP specifically requires that control measures must be implemented if the “sacrifice” (or costs) are not grossly disproportionate to the benefits achieved by the measure. BAT requires that the most effective as well as technically and economically viable techniques are used to provide a high level of protection to the environment as a whole.

The objective of this report is to collate information from the AP1000 NPP design that describes how the environmental protection concept of BAT is applied to prevent or minimise waste emissions and discharges. Much of the information will have previously been presented in the “AP1000 European Design Control Document” [DCD, Reference 3] as ALARA arguments. However, the relevant issues will be brought together to provide information relating the techniques used to prevent or minimise waste emissions and discharges.

3.2 Relationship Between BAT, BPEO and BPM

Best practicable environmental option (BPEO) looks to provide the best strategic option to managing radioactive wastes taking in to account emissions to the environment as a whole (for example, to air, water or land).

Best practicable means (BPM) are the techniques and methods that operators are required to follow under the Radioactive Substances Act 1993 to minimise the volumes and activities of radioactive wastes that are generated and have to be discharged to the environment, and to reduce the impacts of waste management on people and the environment. If the operator is using BPM, radiation risks to the public and the environment will be ALARP.

BPM and BPEO are intended to be applied in fundamentally different ways. Whereas BPEO looks at assessing the best strategic option to apply to managing radioactive wastes, BPM relates to how to optimise the selected option from the perspective of radiological protection. Put simply, BPEO is about doing the right thing and BPM is about doing it the right way. Historically, on UK nuclear licensed sites a BPM study will follow a BPEO study that defines the strategic waste management approach adopted by the operator.

In contrast, BAT is derived from Integrated Pollution Control (IPC) legislation and was originally applied by the agencies for the management of non-radioactive pollutants. Effectively, BPM and BAT are synonymous. Both have the aim of balancing costs against environmental benefits by means of a logical and transparent approach to identifying and selecting processes, operations, and management systems to reduce discharges. It follows that existing BPM studies bear direct relevance to BAT assessments.

4.0 BAT ASSESSMENT – AP1000 NPP NUCLEAR ISLAND

4.1 AP1000 NPP Design

4.1.1 Design Principles – Safety and Simplicity

Westinghouse Electric Company has received standard design certification from the U.S. Nuclear Regulatory Commission (NRC) for the AP1000 NPP design.

The AP1000 NPP design is founded upon rigorously holding to a few inviolate safety principles:

1. No ac power would be required to perform any of the following three key safety functions:

    - Stopping the nuclear reaction
    - Removing the decay heat
    - Maintaining reactor coolant water inventory

    and other safety functions such as:

    - Spent fuel pit cooling
    - Main control room habitability
    - Beyond design basis security related mitigation features.

2. Beyond design basis security-related mitigation features maintain the fission product barriers of the fuel clad, the reactor vessel and coolant system, and the containment vessel. The containment vessel is an ideal barrier against radioactive releases to the environment. Natural, unpumped mechanisms like natural circulation, evaporation, conduction, convection and condensation transfer decay heat out of the core.

3. Minimise core damage frequency and large release frequency as calculated by a robust probabilistic risk assessment (PRA), by designing out failure modes in lieu of designing in mitigation features.

Another underlying philosophy of the AP1000 NPP design process is that the best path to safety is through simplicity. For example, in operating plants today the reactor coolant pumps use a controlled coolant leakage system for establishing a seal on the reactor coolant pump shaft. This shaft seal is a potential source of excessive leakage of reactor coolant. Shaft seal failure mitigation features and safety-related responses to excessive leakage must be provided for these plants. In the AP1000 NPP the shaft seals are eliminated all together through use of canned motor pumps. Another example is the methods of post accident core decay heat removal. Operating plants today use a variety of systems to take reactor coolant out of containment, cool it down, and return it to the core. This creates a large number of potential reactor coolant release scenarios, each requiring a mitigation strategy. In the AP1000 NPP, reactor coolant remains within containment, and only decay heat energy is transferred out of containment. The only remaining containment bypass, reactor coolant release scenarios are the highly unlikely leak in containment itself and the unlikely steam generator (SG) tube leakage event.

In addition to the design objectives of safety first and no ac power for safety-related functions, the AP1000 NPP design process included making constructability, reliability, operability and maintainability part of the design.

Constructability was “designed in” by the use of extensive modularization. Modularization is a technique where portions of the plant are assembled and tested before they are placed in their permanent plant location.

Reliability was “designed in” by using the PRA as a design tool in lieu of simply for design verification. PRA considerations were included in many design decisions. It is important to ensure the lowest public risk.

Operability was “designed in” by designing out the operator for response to design basis accidents. This eliminates the need for operator reliability from the response to accidents. Using natural forces promotes the elimination of the operator from safety-related responses.

Maintainability was “designed in” by extensive layout reviews by both the design team and utility representatives. From the beginning of development, the layout was generated in 3D CAD software. As each item (structure, equipment, pipeline, duct, and tray) was added to the design, it was checked for interferences, inspection access, and maintenance access.

This approach ultimately results in a plant design that is safe because it is simple, and the objectives of lowest hazard to the public and operators, lowest risk, and lowest cost are achieved as by products of the process.

4.1.2 Development of the AP1000 NPP Design

The design of the AP1000 NPP is a development of the AP600™ NPP design. The AP600 NPP design incorporated the simple safety systems evolved for the Secure Military Power Plant (SMPP) originally developed for the United States Air Force. These simple safety systems included a plant driven by natural forces to perform the safety functions that shut down the reactor, keep it cool, and contain its coolant.

The design process used throughout the development of SMPP/AP600/AP1000 is to create a safe nuclear power plant with costs, radiation exposures and radioactive discharges ALARP.

Development of the AP600 was a $450 million design and licensing effort to produce the safest, simplest, least expensive nuclear power plant on the world market. However, where other nuclear plants were not competitive with AP600, other non-nuclear power stations were. In particular, natural gas plants were the economic plants of choice in the U.S. In order to compete against natural gas plant at the time, the AP600 would have to lower its cost per megawatt by over 30 percent. To lower its cost by eliminating any more systems, structures, or components would lessen safety margins and increase risk to the public. Obviously, this approach was rejected. Instead, it was decided to raise the power level of the design without raising the overall plant price an equivalent amount to drive the cost per megawatt down so that a nuclear plant could compete with natural gas plants.

This design power increase needed to be constrained to reap the benefits of the design and licensing effort already invested in the AP600 design. The constraints included:

a. Safety first — maintain large margins to safety limits
b. Maintain passive nature of all safety functions
c. Maintain no operator actions for safety functions
d. Maintain use of proven components and technology
e. Do not change the plant footprint and lose layout and analysis already completed
f. No design impacts unrelated to power
g. Minimise design impacts on the DCD (Design Certification)

The resulting AP1000 NPP design met cost goals while changing only those features necessary to increase power and maintain safety margins. The nuclear island footprint remained unchanged by adding height to the reactor vessel and containment vessel while maintaining their diameters

4.1.3 Key Design Decisions Influencing Environmental Impacts

Over the 15 years of the design life of the AP600, there were many design decisions that reinforced the concept of safety through simplicity, ALARP and BAT [Reference 4]. Examples of the decisions that relate to waste minimisation, waste generation and waste disposal are identified below.

4.1.3.1 Reactor Coolant Pump Selection

The function of the reactor coolant pump is to deliver adequate cooling water for power operations and accident shutdown situations. The classic reactor coolant pump style is a shaft seal pump. It can be made large and can have high hydraulic and electrical efficiencies. However, shaft seals are prone to leakage. Alternatives considered included dc-powered safety pumps, canned motor pumps, no pumps (natural circulation), and others.

Hermetically sealed canned motor pumps were selected based on simplicity and reliability. This decision sacrifices the efficiency of shaft seal pumps for higher inherent reliability and elimination of the potential for reactor coolant leakage from shaft seals. The selection of canned pumps also eliminates the shaft seal pump support systems, such as seal injection, seal leak off, lube oil, and fire protection systems. Unlike shaft seal pumps, canned motor pumps cannot be repaired in situ. The design allows for quick removal and replacement of entire pumps, lowering the hazard and risk to the operators.

A basic premise of the AP1000 NPP design is to maintain safety and respond to accidents without reliance on ac. For post reactor trip core cooling this meant natural circulation through the core to the reactor coolant heat sink. However, relying on natural circulation core cooling in the long term is fine if the core/heat sink thermal centres are far enough apart. Natural circulation does not supply sufficient cooling flow at the very beginning of a shut down transient. The passive solution is the addition of rotating inertia to the canned pumps in the form of a heavy flywheel. The new design features for additional rotating inertia were tested and proven. The pump is not expected to function post accident, and its pressure boundary is continuous without any planned or unplanned leakage.

In summary, the canned motor pump was chosen over the shaft seal pump for reactor coolant service in a process that promoted satisfying its design requirements with lowest radioactive effluent, lowest risk for accidental loss-of-coolant, high reliance on proven technology, lowest risk for public or operator radiation exposure, and lowest overall plant cost.

4.1.3.2 Reactor Coolant Post-LOCA Injection and Cooling

Following a LOCA, reactor plant safety systems must provide makeup for the water lost in such a way as to maintain reactor core cooling. Many PWRs today rely on pumped systems and large sources of water from outside containment to provide this makeup and cooling water. These types of systems require safety grade and seismic Category 1 sources of ac power and water. This approach is counter to the safety with simplicity principle adopted for the AP1000 NPP.

The BAT solution selected for the AP1000 NPP is one that simply relies on total pressure balances and natural circulation. The design focused on developing the simplest set of systems that could maintain core cooling with all safety-related water inventory contained within containment. These include the passive residual heat removal heat exchanger, the core makeup tanks, the accumulators, the in containment refuelling water storage tank, and the passive core cooling long term recirculation system. Safe shutdown conditions with margin, without operator actions, and with no requirement for ac power is achieved.

The benefit of this solution is a very safe, simple set of core cooling features, driven by natural forces, extensively tested and analyzed. It provides safety through simplicity by satisfying its design requirements with no potential radioactive effluent, no risk for accidental loss-of-coolant outside containment, high reliance on proven technology, lowest risk for public or operator radiation exposure, and lowest overall plant cost.

4.1.3.3 Load Follow with Rods

Most central station nuclear power plants today are operated as base load plants. The utilities require that new nuclear plants be designed for a defined level of load follow. To provide some level of load follow today, many plants have systems that manage boron concentrations in and recycle boron in and out of the reactor coolant water. This requires elaborate and complicated boron and water handling systems and results in restrictions on the rate of load follow available.

The AP1000 NPP decision process for load follow control incorporates the proven, safe and simple method of shim rods over the complex method of boron recycle. Mechanical shim control is the use of moveable control rods with low density neutron absorber (gray rods) that can be moved to provide reactivity controls in addition to normal reactivity feedbacks. The gray rod cluster assembly comprises stainless steel rodlets and rodlets containing silver-indium-cadmium absorber material clad with stainless steel. Note that shim rods are used in addition to safety rods and are not needed for reactor shutdown.

The benefit of this solution is that it provides safety through simplicity by satisfying its design requirements with no potential radioactive effluent, no risk for accidental loss-of-coolant outside containment, high reliance on proven technology, lowest risk for public or operator radiation exposure, and lowest overall plant cost while maintaining complete shutdown margin in the shutdown rods.

4.1.3.4 Use of Demineralisers

Radioactive isotopes accumulate in the reactor coolant and spent fuel pool cooling water during operation. Some of these isotopes are gaseous or volatile; most are soluble or suspended in reactor or spent fuel pool coolant water. During plant heat up or coolant boron concentration adjustments by feed and bleed, volumes of this potentially radioactive water accumulate as waste water. In addition, volumes accumulate as a result of sampling operations or as leakage. These sources will accumulate to the point where they must be discharged from the plant. Unlike many plants, the AP1000 NPP uses canned reactor coolant pumps, eliminating the need for shaft seals and their associated potential for reactor coolant leakage from pump shaft seal leak off systems (see subsection 4.1.3.1). In addition, the AP1000 NPP has no plans to recycle dissolved boron in reactor coolant for load follow changes (see subsection 4.1.3.3). By means of these design decisions, the AP1000 NPP radioactive water sources are reduced with the main source coming from letdown during heat up.

The initial BAT decision relates to whether the letdown water during heat up should be stored for reuse during the next cool down. Storage requires additional equipment to store, monitor, process, and recycle relatively small amounts of water. The storage duration could be many months as reuse would only be possible during the next cool down. This approach is unnecessarily complicated, adding radiological hazard risks and additional containment and handling issues. Small amounts of demineralised makeup water are easily added between shutdowns to fulfil the cool down requirements. For these reasons the AP1000 NPP design uses treatment and disposal for the effluent generated by letdown during heat up.

The simple capture of radioactive isotopes in ion exchange resins was chosen over more complicated methods. The design decisions were based on simplicity, reduction of equipment, operations, potential failure modes, and energy loss. The selected process satisfies the design requirements with lowest risk for accidental loss of radionuclides, high reliance on proven technology, and lowest cost.

4.1.3.5 Chemical and Volume Control

The functional requirements for the chemical and volume control system (CVS) are to fill, makeup, letdown, drain, and maintain the proper chemistry of reactor coolant water. In operating plants today, these functions are performed by a variety of safety-related subsystems that are outside containment.

In the AP1000 NPP, the basic design philosophy requires passive systems that eliminate the need for safety-related coolant charging or letdown. The AP1000 NPP design improvements (see subsections 4.1.3.1, 4.1.3.3, 4.1.3.4) have eliminated the requirement to continuously pump borated makeup water into the RCS or to include complicated water processing systems in the design. This allowed additional simplifications to the CVS. The functions of reactor coolant makeup, boron injection, letdown, purification, and others are nonsafety-related making most of the system nonsafety-related. Redundancies and potential safety-related failure modes associated with these functions were eliminated. Boric acid transfer is gravity fed from the boric acid tank to the reactor coolant makeup pump.

In summary, the CVS functional requirements were satisfied by simple designs using a design process that promoted the lowest radioactive effluent, lowest risk for accidental loss-of-coolant, high reliance on proven technology, and lowest risk for public or operator radiation exposure.

4.1.3.6 Post Accident Isotope Control

Radioactive isotopes accumulate in the reactor coolant during operation. During a LOCA, these accumulated isotopes are released into containment. Some of these isotopes are gaseous or volatile; most are soluble or suspended in reactor coolant water. During a LOCA, these soluble and suspended isotopes are dispersed throughout upper containment creating a radiation source. This source can be strong enough to be a hazard to those outside containment.

One option used by many operating plants is use a containment spray system to “wash” these soluble and suspended isotopes out of the containment atmosphere and off the containment walls. These containment spray systems include a water source outside containment, containment penetrations, pumps, valves, nozzles, and other equipment that must be redundant, qualified, controlled, tested, maintained, and repaired.

In the AP1000 NPP natural forces like buoyancy, condensation and conduction move decay heat energy from lower regions of containment to the containment walls. The steam/water mix that condenses on the containment wall returns to the In-containment Refuelling Water Storage Tank (IRWST) or the containment sump by gravity. Through analysis and testing, it has been shown that soluble and suspended isotopes move through upper containment with the water and thus move to the lower portions of containment.

In summary, natural movement of LOCA-related isotopes without containment spray was chosen over a containment spray system in a process that promoted satisfying the AP1000 NPP design requirements with lowest risk for accidental loss-of-coolant, high reliance on proven technology (natural forces), and lowest cost.

4.1.3.7 Beyond Design Basis Features

Beyond design basis features are included in the design to maintain the impact of selected severe accidents to ALARP. The postulated impact of selected severe accidents is calculated using an expanded version of Probabilistic Risk Assessment (PRA) (Level 3). The use of PRA techniques allows for making design decisions in a disciplined way that provides for the lowest risk at the lowest overall incremental cost.

The best example of an AP600 beyond design basis feature is in-vessel retention (IVR). Although numerous PRA techniques are involved in selecting and analyzing beyond design basis events, a common one is core melt. The common design feature for core melt is some form of “core catcher” outside the reactor vessel. A core catcher would have features that precluded recriticality of the corium and cooled it to slow its reaction with materials around the reactor vessel.

In the AP1000 NPP, the need for the core catcher was removed by eliminating the failure mechanism (reactor vessel melt through). In the event of an extremely unlikely severe accident leading to a core melt, the in-containment water sources from the IRWST and passive core cooling components are collected in the lower portions of containment. It is allowed to flow into the reactor vessel insulation structure and next to the reactor vessel. It then cools the reactor vessel by convection and evaporation. The steam rises into the upper containment carrying core heat with it. This steam condenses on the containment vessel inner surface and returns to the lower portion of containment completing the cycle.

In summary, natural movement of in-containment water over the reactor vessel was chosen over a core catcher outside the reactor vessel in a process that promoted satisfying severe accident design requirements with lowest risk for accidental loss of the cooling function, high reliance on proven technology (natural forces), and lowest cost.

4.1.3.8 Selection of Squib Valves

Rapid opening squib valves are used in the AP1000 NPP passive safety systems to isolate the following:

- Stage 4 Automatic Depressurization System (ADS) valves
- IRWST injection line
- Containment recirculation line

These squib valves are used to provide zero leakage during normal operation and reliable opening during an accident. The nature of squib valve body design makes the valve virtually leak free (valve is not subject to internal leakage as with standard valve designs, such as globe, butterfly, gate, check, and the like.). Squib valves are more reliable than air-operated or motor-operated valves because of the reliability of the actuating propellants and the simplicity of the squib valve mechanical design, as compared to other types of valves in the same process application.

4.1.3.9 Low Leakage Containment (passive dose reduction)

Containment is the required last boundary between uncontrolled release of radioactive fission products and the environment. Options include steel containments, concrete containments and steel lined concrete containments. The design requirements for containment include that it must retain gases inside containment up to the containment design pressure, and the design pressure must exceed the maximum expected pressure during a design basis event such as a steam line break or large break LOCA. Another is that the pressure inside containment must be reduced to one half the peak event pressure in 24 hours. There are a number of ways this second requirement has been met, including containment spray and controlled containment leakage or release. However, none of these satisfy the AP1000 NPP requirement of having passive, simple systems.

The designers knew that the simple solution for post accident isotope control was the elimination of containment spray (see subsection 4.1.3.6), and the safest way to control accident releases from containment is not to have any.

The AP1000 NPP has a free-standing steel pressure vessel in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code. This vessel has a high enough design pressure, a large enough free volume, and a large enough heat transfer area to accommodate the worst design basis pressure challenge without the requirement to vent. Pressure vessel design requirements extend to all penetrations and attachments. The addition of passive containment cooling by distributing water over the exterior of the vessel provides a passive means of aiding heat removal and reducing internal pressure. This solution was chosen using a process that promoted satisfying design requirements with the simplest possible design goal, eliminating the likelihood of containment leakage or the need for containment venting.

4.1.3.10 Catalytic Hydrogen Recombiner (passive)

There are a variety of mechanisms in a nuclear power plant that can generate free hydrogen gas. Most of these generate very small amounts, while some relating to beyond the design basis severe accidents can generate large amounts. Regardless of the source, accumulations of hydrogen can rise to a potentially explosive level. To ensure continuous, simple, hydrogen removal capability that does not rely on ac power and can be environmentally qualified for post accident service, catalytic hydrogen recombiners were chosen for in-containment hydrogen control. These recombiners are in addition to the hydrogen igniters placed throughout containment.

The catalytic hydrogen recombiners were chosen over more complicated hydrogen removal schemes in a process that promoted satisfying design requirements with lowest radioactive effluent, lowest risk for hydrogen burning, high reliance on proven technology, and lowest risk for public or operator radiation exposure.

4.1.3.11 Trisodium Phosphate Baskets (passive)

Post-LOCA conditions within containment require that the free water in containment be treated to maintain its pH within prescribed limits. This is done to ensure the optimum chemical speciation of the fission products. In many operating plants, this pH control is established by the chemistry of the containment recirculation water brought in from tanks outside containment.

In the AP1000 NPP, two passive options with no ac requirements were considered. These were in-containment tanks with buffer solution and safety-related controls, and in-containment baskets with solid trisodium phosphate. The solution chosen was to install baskets low in containment that hold solid trisodium phosphate. In the event of a LOCA, the water accumulating in lower region of containment would self-buffer by dissolving the chemical.

This solution benefits from being a very safe, simple post LOCA in-containment pH buffering system that is driven by natural forces and has been extensively tested and analyzed. It provides safety through simplicity by satisfying its design requirements with no potential radioactive effluent, no risk for accidental loss-of-coolant outside containment, high reliance on proven technology, lowest risk for public or operator radiation exposure, and lowest overall plant cost.

4.1.3.12 In-containment CVS System

One of the functional requirements for the CVS is to maintain the proper chemistry of reactor coolant water. This includes removal of impurities (both radioactive and non-radioactive) from the RCS. In operating plants today, this function is performed by taking a portion of the reactor coolant out of containment, reducing its pressure and temperature, purifying it, and forcing it back into containment and the RCS with a high pressure pumping system. This process introduces potential reactor coolant leak sites outside containment, as well as imposing additional reactor coolant inventory control requirements. Since the tradeoffs to simplicity in other portions of the CVS have been made (see subsections 4.1.3.1, 4.1.3.3, 4.1.3.4 and 4.1.3.5), there is no other reason to continuously pump makeup water into the RCS.

A simple approach to coolant purification was developed that performed continuous purification of a portion of the reactor coolant at reactor coolant pressure, using reactor coolant pump head as a motive force and keeping all the purification equipment and reactor coolant within the containment vessel. High pressure water purification using ion exchangers is an industry proven process.

In summary, the in-containment, high pressure coolant purification was chosen over out of containment, pumped, low pressure purification. This created a process that promoted satisfying design requirements with lowest radioactive effluent, lowest risk for accidental loss-of-coolant, high reliance on proven technology, and lowest risk for public or operator radiation exposure.

4.1.3.13 Zinc Addition

Chemical build-up in the RCS has the potential to cause water stress corrosion cracking and crud induced power shift. The AP1000 NPP CVS incorporates a zinc acetate addition subsystem to reduce these effects. Zinc concentrations ranging from 5 to 40 parts per billion (ppb) in the RCS change the oxide film on primary piping and components that significantly reduces the potential for these adverse impacts. Zinc addition has also been found to significantly reduce occupational radiation exposure by as much as 50 percent when incorporated as early as hot functional testing.

4.1.3.14 Reduction of Containment Penetrations

Penetrations through the containment are designed to be leak tight assemblies, allowing pipes and cables to pass through the leak tight containment vessel boundary. Very often, they are the sites of small leak paths.

One of the fundamental design objectives for passive cooling of the AP1000 NPP is to isolate containment during a design basis accident with no ac supply, so that only energy passes through the containment boundary, not fluids. This minimises the number of penetrations and reduces design, inspection, and maintenance burdens.

Designers further reduced penetrations by implementation of a variety of innovative techniques. Service systems in containment like component cooling water or compressed air are split and routed inside containment resulting in only one supply or return penetration for each service. Some intermittent services with common fluids share common penetrations. For example, both chilled water and hot water heating services to heating, ventilation and air conditioning (HVAC) in containment share common penetrations since they will not be used at the same time. Instrumentation and control penetrations are reduced by taking advantage of digital data highway technology. Multiplexing cabinets are located such that instrumentation and control signals share a common highway penetration in lieu of multiple individual signal penetrations.

This solution was chosen using a process that promoted satisfying design requirements with lowest number of containment penetrations, high reliance on proven technology, lowest risk for containment leakage, and public or operator radiation exposure.

4.1.3.15 Air Diaphragm Waste Pumps

Liquid waste water (oily, radioactive, non-radioactive) must be transferred within the plant from tank to tank or for processing, and must be transferred out of the plant. In plants today this transfer is powered by a wide variety of pump types (centrifugal, positive displacement, air operated, and others). The tradeoff was to continue with this variety approach or pick a standard pump type for all AP1000 NPP waste pump services.

After consideration of the available types, the decision was made to use inexpensive, simple, air operated, fully contained pumps for waste water service. In these types of pumps the working fluid remains inside its pressure boundary. This eliminates any chance of seal leakage since there are no seals, especially no rotating seals.

The benefit of this solution is a very safe, simple set of pumps, common for common service. It provides safety through simplicity by satisfying design requirements with no potential radioactive or oily effluent, no risk for accidental loss of radioactive fluid outside containment, high reliance on proven technology, lowest risk for public or operator radiation exposure, and lowest overall plant cost.

4.1.3.16 Use of Digital Instrumentation and Control

The AP1000 NPP incorporates digital instrumentation and control. They offer improved process control, reliability and availability improvements, and significantly reduced cost for operation and maintenance. The use of a data highway eliminates large quantities of mechanical instrumentation and control components, cabling, cable tray, cable spreading areas, containment penetrations and other equipment. It provides a safe, simple platform for plant protection and control.

The benefits listed above result in a more reliable, efficient, and modern plant.

4.1.3.17 Use of Advanced Control Room

The control room is the main focal point for the safe monitoring and control of the AP1000 NPP plant design. The use of modern operator-interface technology, in the main control room

(MCR) represents a move away from the traditional "control board" control room design. The amount of fixed controls and displays has been minimised to the extent practical. The main operator-machine interface is via computer-based monitors, mouse and keyboards. The visual display unit-based (VDU-based) operator-interface integrates a number of systems into one flexible interface technology. This includes the use of large screen displays that enables plant overview and alarm status information to be visible from any likely operator location in the MCR. This facilitates crew group plant status awareness and decision-making.

The current technology has been proven to improve operator performance, increase productivity, and reduce the likelihood of human errors by using safe, simple technology. Furthermore, the technology enables a decrease in the number of operations personnel required in the control room and assists in reducing electric generation costs.

4.1.3.18 Resistance to Airplane Crash

The construction of the shield building of the AP1000 NPP has been upgraded from the U.S. concrete design of the AP600 to a plate and concrete sandwich structure to satisfy the U.S. design requirements for resistance to terrorist attack by commercial aircraft. The ventilation system has also been modified to replace the 15 large ventilation inlets of the AP600 with 384 small inlet ducts in the AP1000 NPP. The smaller inlets consist of square steel tubes inclined upward from the outside face to the inside face. The redesigned air inlets also provide a significant increase in shield building resistance to restrict debris or fuel from entering the building due to their small size and orientations. The redesign of the air inlets provide inherent protection against aircraft impact while maintaining the design functions associated with provided passive containment cooling. These solutions were chosen because of improved performance in meeting more stringent design specifications, high reliance on proven technology, and lowest risk for public or operator radiation exposure.

4.2 AP1000 NPP Emissions and Discharges

The AP1000 NPP radioactive waste (radwaste) management systems control the handling and treatment of liquid, gaseous, and solid radwaste. These systems include the liquid radwaste system (WLS), the gaseous radwaste system (WGS), and the solid radwaste system (WSS).

4.2.1 Air Emissions

4.2.1.1 Radioactive Air Emission Sources

During reactor operation, tritium, C-14 and radioactive isotopes of xenon, krypton and iodine are created as fission products. A portion of these radionuclides is released to the reactor coolant because of a small number of fuel cladding defects. Tritium is also produced by neutron activation reactions of boron, lithium, or deuterium occurring in the primary coolant. Trace quantities of activated corrosion products can also accumulate in the primary coolant. Leakage of reactor coolant thus results in a release to the containment atmosphere of the radioactive gases and activated corrosion products released as aerosols and particulates. Activation of naturally occurring Ar-40 and N-14 in the containment atmosphere close to the reactor vessel also occurs to produce radioactive Ar-41 and C-14, respectively.

4.2.1.2 Non-radioactive Air Emission Sources

The AP1000 NPP design does not incorporate an incinerator. The non-radioactive air emission sources are mainly associated with emissions from the standby diesel generators. These generators have a thermal rated input of ~13MW each and, as such, are below the threshold of 20MW for regulated combustion devices. The combustion gas emissions (SO_2,

NO_x CO and particulates) are not significant because of the small size of the generators and because the generators will not be in normal use.

4.2.1.3 Air Release Pathways

The radioactive emissions are released to atmosphere by way of the following pathways.

4.2.1.3.1 Plant Vent

The plant vent provides a radiation monitored release path for the following:

- Containment venting releases

  The containment contains activity as a result of leakage of reactor coolant and as a result of activation of naturally occurring Ar-40 in the atmosphere to form radioactive Ar-41. The containment purge is vented to the plant vent.

- Auxiliary/annex building ventilation releases

  The auxiliary/annex building ventilation subsystem serves radiologically controlled equipment, piping and valve rooms and adjacent access and staging areas. These areas could contain activity as a result of leakage from process streams.

- Radwaste building releases

  The radwaste building contains and processes radioactive waste streams. This area could contain activity as a result of leakage from processing the waste.

- Gaseous radwaste system discharge

  The WGS is designed to receive hydrogen bearing and radioactive gases generated during process operation. The radioactive gas flowing into the WGS enters as trace contamination in a stream of hydrogen and nitrogen.

4.2.1.3.2 Turbine Building Vent

The turbine steam sealing (gland seal) system exhaust and the condenser air removal system exhaust, which includes the gland seal exhaust during plant startup, are routed to a common header that discharges the exhausts to the environs via a radiation-monitored turbine building vent. The gland seal system and condenser air removal system exhausts are not filtered prior to their release to the environs, as they are not normally radioactive. The turbine island vent discharge radiation monitor measures the concentration of radioactive gases in the steam and non-condensable gases that are discharged by the condenser vacuum pumps and the gland seal steam condenser. This measurement provides early indication of leakage between the primary and secondary sides of the steam generators. The monitor provides an alarm in the MCR if concentrations exceed a predetermined setpoint. Upon detection of unacceptable levels of radiation in the exhausts, which may occur as a result of a steam generator tube leak, appropriate corrective actions will be manually performed.

The exhaust ducts from the battery rooms are also connected to the turbine building vent to remove hydrogen gas generated by the batteries.

4.2.1.3.3 Building Ventilation Systems

The air conditioning, heating, cooling, and ventilation system comprise a number of systems that serve the various buildings and structures of the plant (see Table 4-1).

4.2.1.4 Air Emissions from Normal Operations

The WGS periodically receives influent when CVS letdown is processed through the WLS degasifier during RCS dilution and volume control operations.

The other major source of input to the WGS is the reactor coolant drain tank. Hydrogen dissolved in the influent to the reactor coolant drain tank enters the WGS either via the tank vent or the WLS degasifier discharge. The tank vent is normally closed, but is periodically opened on high pressure to vent the gas that has come out of solution. The reactor coolant drain tank liquid is normally discharged to the WLS via the degasifier, where the remaining hydrogen is removed. The reactor coolant drain tank is purged with nitrogen gas to discharge nitrogen and fission gases to the WGS before operations requiring tank access. The reactor coolant drain tank is also purged with nitrogen gas to dilute and discharge oxygen after tank servicing or inspection operations which allow air to enter the tank.

The radioactive emissions from the AP1000 NPP that are expected throughout the 18-month fuel cycle are shown in Table 4-2.

In general, there is a slight increase in emissions over the fuel cycle as fuel defects increase. The normal operational emissions occur between months 1-16. In months 17-18, emissions increase further due to preparations for fuel replacement.

4.2.1.5 Air Emissions from Non-routine Operations

Removal of radioactive gases from the RCS is not normally necessary because the gases do not build up to unacceptable levels when fuel defects are at or below the design basis level of 0.25%. If radioactive gas removal is required because of high fuel defects, the CVS can be operated by routing flow to the WLS degasifier.

Removal of radioactive gas and hydrogen during shutdown operations is necessary to avoid extending the maintenance and refuelling outages. The RCS pressure boundary cannot be opened to the containment atmosphere until the gas concentrations are reduced to low levels. The gaseous emissions expected during refuelling operations are shown in months 17-18 of Table 4-2.

The turbine building vents provide the release path for the condenser air removal system, gland seal condenser exhaust and the turbine building ventilation releases. These emissions are potentially radioactive following leakage between the primary and secondary sides of the steam generators. However, such leaks will be detected by radiation monitoring devices in the duct allowing corrective action to be initiated.

4.2.2 Water Discharges

4.2.2.1 Sources

4.2.2.1.1 Radioactive Liquid Waste

Fission reactions produce gaseous radioactive isotopes (see subsection 4.2.1.1) and also isotopes of strontium (for example, Sr-90), iodine (for example, I-133) and caesium (for example, Cs-137). A portion of these radionuclides is released to the reactor coolant because of a small number of fuel cladding defects. Trace quantities of activated corrosion products can also accumulate in the primary coolant.

During a reactor heatup, it is necessary to remove reactor coolant due to expansion. This is carried out via the CVS. The CVS is also used to adjust the concentration of boron in the RCS, as required, to maintain the desired control rod position with core depletion. RCS boron changes are required to compensate for fuel depletion, startups, shutdowns, and refuelling. The reactor coolant removed by the CVS becomes radioactive liquid waste that must be treated before disposal.

Leakage of reactor coolant can also result in release of radioactive effluent. Radioactivity can also enter the secondary coolant systems from steam generator tube leakage. Contaminated secondary coolant is also designated as radioactive liquid waste.

4.2.2.1.2 Non-radioactive Liquid Waste

The non-radioactive liquid waste sources arise from routine discharges, blowdown or leakage from non-radioactive water systems (see Table 4-3).

4.2.2.2 Release Pathways

Liquid waste is collected in one of the following systems:

- Radioactive Drains

  The radioactive waste drain system is arranged to receive inputs from the radiologically controlled areas of the auxiliary, annex, and radwaste buildings based on segregation of the liquid wastes into chemical and non-chemical drains. The radioactive waste drain system collects radioactive liquid wastes at atmospheric pressure from equipment and floor drainage of the radioactive portions of the auxiliary building, annex building, and radwaste building and directs these wastes to a centrally located sump located in the auxiliary building. The contents of the sump are pumped to the WLS tanks.

- Chemical Waste Drains

  The radioactive waste drain system collects chemical wastes from the auxiliary building chemical laboratory and decontamination solution drains from the annex building and directs these wastes to the chemical waste tank of the WLS.

- Detergent Waste Drains

  The laundry and respirator cleaning functions that generate detergent wastes are performed offsite. Detergent wastes from hot sinks and showers are routed to the chemical waste tank.

- Non-radioactive and Potentially Radioactive Drains

  The waste water system collects non-radioactive waste from floor and equipment drains in auxiliary, annex, turbine, and diesel generator building sumps or tanks. Selected normally non-radioactive liquid waste sumps and tanks are monitored for radioactivity to determine whether the liquid wastes have been inadvertently contaminated. If contaminated, the wastes are diverted to the WLS for processing and ultimate disposal.

- Oily Waste Drains

  The waste water system collects non-radioactive, oily, liquid waste in drain tanks and sumps. Drain tank and sump liquid wastes are pumped through an oil separator prior to further processing. The oil is collected in a tank for disposal. Sampling for oil in the waste holdup tank of the WLS is provided to detect oil contamination before the ion exchanger resins are damaged. Oily water is pumped from the tank through an oil adsorbing bag filter before further processing. The spent bag filters are transferred to drums, stored in the radwaste building and monitored for activity before determining appropriate off-site location.

Radioactive effluent at a normal flow rate of 8 m^3/day is treated in the liquid radwaste plant and collected in monitor tanks. If the effluent is below the discharge limits, it is released to the environment by blending with the seawater cooling return. Non-radioactive liquid waste passes through an oil water separator before discharge via the cooling water system.

4.2.2.3 Radioactive Effluent Arising from Normal Operations

Radioactive water discharges from normal operations include effluent arising from the reactor coolant drain tank and the CVS. The reactor coolant drain tank collects drainage from various primary systems and components inside containment. Effluent from the CVS is produced mainly as a result of RCS heatup, boron concentration changes and RCS level reduction for refuelling.

The radioactive effluent discharges following treatment in the WLS are shown in Table 4-4 for the 18-month fuel cycle.

In general, there is a slight increase in discharges over the fuel cycle as fuel defects increase. The normal operational emissions occur between months 1-16. In months 17-18 emissions increase further due to preparations for fuel replacement.

4.2.2.4 Radioactive Effluent Arising from Non-routine Operations

Non-routine sources of radioactive liquid effluent include leakage from various primary systems and components inside containment and contaminated water on the secondary side following from steam generator tube leakage.

4.2.2.5 Non-radioactive Effluent Arising from Normal Operations

Normal operations result in sanitary waste water, cooling water and storm water runoff discharges. Normal process operations also produce waste water from processes equipment and floor drains from non-radioactive buildings. The non-radioactive liquid effluents are identified in Table 4-5.

4.2.3 Solid Wastes

4.2.3.1 Radioactive Solid Wastes

Solid radioactive wastes comprise LLW, ILW, and high level waste (HLW).

LLW is radioactive material that is not acceptable for disposal with ordinary refuse. LLW is required to be controlled for the protection of people; however, LLW does not require shielding during handling and storage processes. These include general and mixed wastes as a result of normal plant operation such as refuse bins, personal protective equipment (PPE), wipes and other consumables. The LLW solid waste sources are summarized in Table 4-6 and detailed in Section 3.5 of the "UK AP1000 Environment Report" [Reference 5].

ILW is material with higher radioactivity than LLW. Handling and storage of ILW require shielding. ILW is categorised into dry ILW and wet ILW. Dry ILW consists of spent filter cartridges and miscellaneous contaminated plant items (for example, broken valves) from the AP1000 NPP and surrounding auxiliary buildings. Wet ILW consists of spent ion exchange resins and deep bed filtration media from the AP1000 NPP reactor building. The ILW solid waste sources are summarized in Table 4-6 and detailed in Section 3.5 of the "UK AP1000 Environment Report" [Reference 5].

The HLW is related to spent fuel and its handling and disposal is covered Section 3.5 of the "UK AP1000 Environment Report" [Reference 5]. The volumes generated are summarized in Table 4-6.

Waste generated during decommissioning activities is described in subsection 3.5.10 of the "UK AP1000 Environment Report" [References 5 and 6].

4.2.3.2 Non-radioactive Solid Wastes

The non-radioactive solid wastes are identified in Table 4-7.

4.3 AP1000 NPP Emission Control Techniques

4.3.1 Minimisation at Source

Minimisation of emissions at source is a principle that is consistent with the objectives of BAT. In the AP1000 NPP there are several ways in which the release of radioactive emissions to atmosphere is reduced at source:

4.3.1.1 Fuel Rod Burnup

The fuel economics and the amount of spent fuel are closely correlated. Both are optimized when the fuel cycle is designed with fuel being discharged from the reactor as close as is reasonable to the licensed discharge burnup limit. The current licensed limit for Westinghouse fuel is 62,000 MWD/MTU on the lead rod maximum burnup. Typically, a batch average burnup of approximately 50,000 MWD/MTU is achieved.

4.3.1.2 Operational Cycle

Utilities can operate the AP1000 NPP on many different cycle lengths (for example, annual vs. 18-month cycles) as best meets their operational needs. If the prime objective is to reduce the average number of discharge assemblies per year, then an annual cycle in the AP1000 NPP would discharge fewer assemblies on the average than an 18-month cycle (40 vs. 43). However, depending on the cost of the extra outage every three years, combined with the cost of replacement power during the outage, the impact of outage length on average capacity factor, and the like, this may not be the most overall economically efficient operation of the core. The vast majority of Westinghouse customers choose the longer 18-month fuel cycle.

Spent fuel issues are discussed in detail in Section 3.5 of the "UK AP1000 Environment Report" [Reference 5].

4.3.1.3 Fuel Rod/Cladding Design

The AP1000 NPP fuel rods consist of cylindrical, ceramic pellets of slightly enriched uranium dioxide (UO_2). These pellets are contained in cold-worked and stress-relieved ZIRLO® tubing, which is plugged and seal-welded at the ends to encapsulate the fuel. Sintered, high-density uranium dioxide fuel reacts only slightly with the clad at core operating temperatures and pressures. In the event of clad defects, the high resistance of uranium dioxide to attack by water protects against fuel deterioration, although limited fuel erosion can occur. The consequences of defects in the clad are greatly reduced by the ability of UO_2 to retain fission products, including those which are gaseous or highly volatile. ZIRLO® is an advanced zirconium-based alloy which has a high corrosion resistance to coolant, fuel, and fission products; and high strength and ductility at operating temperatures. Selection of ZIRLO® cladding materials minimises the formation of defects that can result in radioactive releases to the reactor coolant.

4.3.1.4 Materials Selection

Hardfacing material in contact with reactor coolant is primarily a qualified low or zero cobalt alloy equivalent to Stellite-6. The use of cobalt base alloy is minimised. Low or zero cobalt alloys used for hardfacing or other applications where cobalt alloys have been previously used are qualified using wear and corrosion tests. The corrosion tests qualify the corrosion resistance of the alloy in reactor coolant. Cobalt-free wear resistant alloys considered for this application include those developed and qualified in nuclear industry programs.

The parts of the control rod drive mechanisms and control rod drive line exposed to reactor coolant are made of metals that resist the corrosive action of the coolant. Three types of metals are used exclusively: stainless steels, nickel-chromium-iron alloys, and, to a limited extent, cobalt-based alloys. These materials have provided many years of successful operation in similar control rod drive mechanisms. In the case of stainless steels, only austenitic and martensitic stainless steels are used. Cobalt-based alloys have limited use in the AP1000 NPP design. Where low or zero cobalt alloys are substituted for cobalt-based alloy pins, bars, or hard facing, the substitute material is qualified by wear and corrosion tests.

4.3.1.5 Control of Reactor Coolant Water Chemistry

The RCS contains boric acid for long-term reactivity control of the core. The RCS water chemistry is controlled to minimise corrosion by the addition of chemicals using the chemical and CVS. Lithium hydroxide (Li_7OH) is used to control the pH of the RCS and minimise the formation of tritium. Li_7OH is chosen for its compatibility with the material and water chemistry of borated water, stainless steel, and zirconium systems. During plant startup from cold shutdown, hydrazine is introduced as an oxygen scavenger. During power operations, dissolved hydrogen is added to the RCS to eliminate free oxygen produced by radiolysis in the core and to prevent ammonia formation. The RCS water chemistry is routinely analyzed to ensure that the chemistry is correct and corrosion product particulates are below specified limits.

4.3.1.6 Gray Rods and Burnable Absorber Rods

Core reactivity is controlled by means of a chemical poison (boric acid) dissolved in the coolant, rod cluster control assemblies, gray rod cluster assemblies, and burnable absorbers.

The gray rod cluster assemblies are used in load follow manoeuvering and provide a mechanical shim reactivity mechanism which eliminates the need for chemical shim control provided by changes to the concentration of soluble boron.

Discrete burnable absorber rods or integral fuel burnable absorber rods or both may be used to provide partial control of the excess reactivity available during the fuel cycle. In doing so, the burnable absorber rods reduce the requirement for soluble boron in the moderator at the beginning of the fuel cycle.

The reactor controls provided by gray rods and burnable absorber rods reduce the requirements for varying the boron concentrations in the RCS. By doing so the volume of reactor coolant that is withdrawn by the CVS and treated in the WLS is reduced.

4.3.1.7 Reactor Coolant Pressure Boundary

Airborne releases can be limited by restricting reactor coolant leakage. The reactor coolant pressure boundary (RCPB) provides a barrier against the release of radioactivity generated within the reactor. The RCPB comprises the vessels, piping, pumps, and valves that are part of the RCS, or that are connected to the RCS up to and including the following:

- The outermost containment isolation valve in system piping that penetrates the containment
- The second of two valves closed during normal operation in system piping that does not penetrate containment
- The RCS overpressure protection valves

The RCPB is designed to contain the coolant under operating temperature and pressure conditions and limit leakage (and activity release) to the containment atmosphere. RCPB leakage detection is accomplished by diverse measurement methods, including level, flow, and radioactivity measurements. Monitoring provides a means of detecting and to the extent practical, identifying the source and quantifying the reactor coolant leakage.

4.3.1.8 Reactor Coolant Purification

The CVS purifies the RCS to maintain low RCS activity levels. The CVS purification loop contains two mixed bed demineralisers, an optional cation bed demineraliser, and two reactor coolant filters. The mixed bed demineralisers are provided in the purification loop to remove ionic corrosion products and certain ionic fission products. The demineralisers also act as filters. The reactor coolant filters are provided downstream of the demineralisers to collect particulates and resin fines.

One mixed bed is normally in service; with a second demineraliser acting as backup in case the normal unit should become exhausted during operation. Each demineraliser and filter is sized to provide a minimum of one fuel cycle of service with change-out of the in-service demineraliser normally occurring at the end of each fuel cycle, irrespective of the conditions and chemical exposure history during the fuel cycle. Unforeseen or unexpected events or transients in contaminant loading could potentially necessitate the premature need to remove the primary CVS purification mixed bed from service. In this case, the back-up CVS mixed bed can be placed in service without the need to enter containment. At that point, it would be left to the judgment of the operating utility whether there is a pressing need to replace the exhausted CVS bed. The radiological conditions during change-out must be carefully assessed, particularly if change-out is required with the unit in power operation.

The CVS Mixed Bed Demineralisers have limited capability for deboration. The purification mixed bed that is in service at any given time will already be operating fully equilibrated with boron. The designated "back-up" CVS Mixed Bed Demineraliser (not yet in service) has the capability to perform deboration of roughly 70 ppm boron at the end of the fuel cycle. If the back-up mixed bed is used only for end of cycle deboration of the RCS, then that mixed bed may be suitable for use as the purification mixed bed in the following fuel cycle. This approach minimises the generation of ILW.

The mixed bed demineralisers also remove zinc during periods of zinc addition (see Section 4.1.3.13). Approximately 8% of the mixed bed cation resin sites may be converted to the zinc form following 18 months of continuous CVS mixed bed operation at 10 ppb zinc in the RCS.

The mixed bed demineraliser in service can be supplemented by intermittent use of the cation bed demineraliser for additional purification in the event of fuel defects. In this case, the cation resin removes mostly lithium and caesium isotopes. The cation bed demineraliser has sufficient capacity to maintain the caesium-136 concentration in the reactor coolant below 1.0 $\mu Ci/cm^3$ with design basis fuel defects. Each mixed bed and the cation bed demineraliser is sized to accept the maximum purification flow.

The CVS ion exchange treatment also removes radioactive iodine concentrations in the reactor coolant. Removal of the noble gases from the RCS is not normally necessary because the gases will not build up to unacceptable levels when fuel defects are within normally anticipated ranges. If noble gas removal is required because of high RCS concentration, the CVS can be operated in conjunction with the liquid radwaste system degasifier, to remove the gases.

By maintaining low RCS activity levels, the radioactive releases associated with reactor coolant leakage to the containment atmosphere is reduced.

4.3.1.9 Recycling Steam Generator Blow Down

Fluid recycling is provided for the steam generator blowdown fluid which is normally returned to the condensate system.

4.3.2 Air Treatment Systems

4.3.2.1 Gaseous Radwaste System

The WGS controls, collects, processes, stores, and disposes of gaseous radioactive wastes generated during normal operation, including anticipated operational occurrences. The WGS involves the gaseous radwaste system, which deals with potentially hydrogen-bearing and radioactive gases generated during plant operation. Additionally, it involves the management of building ventilation, containment purge, and condenser air removal system exhausts.

The major inputs to the WGS are RCS gases stripped from the CVS letdown flow by the WLS vacuum degasifier during RCS dilution and boration, as well as during degassing prior to a reactor shutdown. Other inputs to the WGS are the gases from the reactor coolant drain tank vent and the gases stripped from the reactor coolant drain tank liquid by the WLS degasifier. The WGS system is not normally in operation. It is operated, as necessary, when the above operations are carried out. The design basis period of operation is the last 45 days of a fuel cycle which is when the RCS dilution and subsequent letdown from the CVS into the WLS is at a maximum. During these 45 days, approximately 26 m^3 of predominantly hydrogen and nitrogen (as carrier gases), fission gases, and water vapor are released.

The treatment is a once-through, ambient-temperature, activated carbon delay system comprising the following four stages:

- Gas Cooler

  This cools the influent waste gas to 4.4°C (40°F) by a chilled water system. The pressure of the gas flow through the gas cooler is less than the chilled water pressure to minimise the potential for contaminating the chilled water system.

- Moisture Separator

  This removes the moisture formed when the gas steam is cooled. The moisture separator is sized for the design basis purge gas flow rate and is oversized for the lower normal flow rate.

- Guard Bed

  The activated carbon guard bed protects the delay beds from abnormal moisture carryover or chemical contaminants by removing them from the waste stream. Under normal operating conditions, the guard bed provides increased delay time for xenon and krypton, and removes iodine entering the system. The flow through the activated carbon bed is downward. A retention screen on the outlet of the guard bed prevents the loss of activated carbon from the unit.

- Activated Carbon Delay Beds

  Two activated carbon delay beds in series are provided where the release of xenon and krypton is delayed by a dynamic adsorption process. During the delay period the radioactive decay of the fission gases significantly reduces the radioactivity of the gas flow leaving the system.

  The minimum calculated holdup times are 38.6 days for xenon and 2.2 days for krypton, based upon a continuous input flowrate to the WGS of 0.5 scfm (0.85 m^3/h). However, the design basis period of operation is the last 45 days of a fuel cycle when the RCS dilution and subsequent letdown from the CVS into the WLS peaks. During this period, the WGS input is at a maximum and the average input flowrate is 0.014 scfm (0.024 m^3/h), which results in longer hold up times being achieved.

  The two beds together provide 100 percent of the stated system capacity under design basis conditions. During normal operation a single bed provides adequate performance. This provides operational flexibility to permit continued operation of the WGS in the event of operational upsets in the system that requires isolation of one bed. Normal operation will be with two beds in series and it is not expected that a delay bed will be out of service on a frequent basis.

  No final filter is incorporated in the WGS because the carrier-gas velocity through the beds is very low, and flow in the final leg of the delay beds is oriented upward through the bed. Therefore, the potential for particulate carry-over is not judged to be significant, and the complexity associated with an outlet filter is not justified.

The system contains provisions for continuously monitoring the moisture level at the inlet of the guard bed. Monitoring the performance of individual components in the WGS is done by collecting and analyzing grab samples. Connections between the two delay beds allow for the collection of samples at the inlet and outlet of the guard bed, and at the outlet of the second delay bed.

The WGS has a radiation monitor that continuously monitors the discharge from the delay beds. The monitor will automatically send a signal to terminate the discharge when the radioactive waste management radiation level in the discharge stream reaches a predetermined setpoint.

The effluent from the WGS is discharged to the environs via the system ventilation exhaust duct connected to the plant vent.

Compared to alternative, compressed gas storage type gaseous radwaste designs, the charcoal delay bed provides simplicity and reliability of operation and typically provides effective control of off-site doses while minimising operational radiation exposure. In addition, the added complexity of the historic compressed storage system increases the probability of operator error resulting in a premature venting and release of a decay tank, which would tend to increase releases. This error mode is eliminated with the more passive charcoal delay bed system of the AP1000 NPP.

4.3.2.1.1 BAT – Optimisation of Delay Bed Sizing

The carbon delay beds have been designed as a folded serpentine configuration to minimise space requirements and the potential for voids in the activated carbon. The length-to-diameter ratio will maximize the ratio of breakthrough time to mean delay time. The waste gas flow is generally vertical (up and down) through columns of granular activated carbon. No retention screens are required on the delay bed since the flow is low velocity and enters and leaves each delay bed at its top.

Each serpentine has four legs. The number of legs, and hence the volume of carbon in the delay bed has been optimized by evaluating the radioactive releases (using the GALE code) expected as a function of the number of legs. Figure 4-1 shows how the optimum number of legs in the delay bed system is eight. Increasing the number of legs above eight has a diminishing benefit in terms of reducing releases of radioactivity. Increasing the size of the delay bed is not warranted in terms of the cost of increasing volumetric space requirements within the auxiliary building, which is a seismic Category 1 building; the cost of purchase, installation, and decommissioning of the additional serpentine legs; and the additional cost of activated carbon.

The charcoal beds are essentially passive in nature and do not typically require maintenance.

4.3.2.2 Building Ventilation Systems with Abatement

4.3.2.2.1 Containment Air Filtration System

The containment building can contain activity as a result of leakage of reactor coolant and as a result of activation of naturally occurring Ar-40 in the atmosphere to form radioactive Ar-41.

The Containment Air filtration System VFS purges the containment by providing fresh air from outside and exhausting air to the plant vent. The air exhausted by the VFS is filtered with high efficiency filters, charcoal filters and post filters. The VFS also exhausts from areas served by the Radiologically Controlled Area Ventilation System (VAS) and the Health Physics and Hot Machine Shop HVAC System (VHS) after receipt of a High radiation signal in the VAS or the VHS exhaust respectively.

The VFS comprises two parallel systems which may be operated individually or simultaneously as required by the operating regime with or without associated inlet air handling units. The two exhaust air filtration units are located within the radiologically controlled area of the annex building. Each exhaust air filtration unit can handle 100% of the system capacity. The VFS system is diesel backed to improve its reliability.

Each VFS unit consists of an electric heater, an upstream high efficiency particulate air (HEPA) filter bank, a charcoal adsorber with a downstream post-filter bank, and an exhaust fan. The efficiencies of the individual filtration elements are in Table 4-8. A gaseous radiation monitor is located downstream of the exhaust air filtration units in the common ductwork to provide an alarm if abnormal gaseous releases are detected.

During normal plant operation, the VFS operates on a periodic basis to purge the containment atmosphere as determined by the MCR operator to reduce airborne radioactivity or to maintain the containment pressure within its normal operating range.

The filtered exhaust air from the containment is discharged to the atmosphere through the plant vent by the VFS exhaust fan. Radioactivity indication and alarms are provided to inform the MCR operators of the concentration of gaseous radioactivity in the VFS exhaust duct. There are additional VFS radiation monitors that measure gaseous, particulate, and iodine concentrations in the plant vent.

4.3.2.2.2 Radiologically Controlled Area Ventilation System

The Radiologically Controlled Area Ventilation System (VAS) serves the radiologically controlled areas of the auxiliary and annex buildings. The VAS consists of two separate once through type ventilation subsystems; the auxiliary/annex building ventilation subsystem and the fuel handling area ventilation subsystem. It is considered that there is a lower risk of radiological contamination in the general area served by the auxiliary/annex building ventilation subsystem than the fuel handling area. As a result the design of these ventilation subsystems is different.

The fuel handling area ventilation subsystem supply and exhaust ductwork is arranged to exhaust the spent fuel pool area separately from the auxiliary building. It provides directional airflow from the rail car/bay filter storage area into the spent resin equipment rooms. The exhaust fans normally pass the exhaust air through a HEPA filter system at an approximate flowrate of 5.52 m^3s^{-1} (11,700 cfm) before discharge via the plant vent.

The auxiliary/annex building ventilation subsystem is routed to minimize the spread of airborne contamination by directing the supply airflow from the low radiation access areas into the radioactive equipment and piping rooms with a greater potential for airborne radioactivity. Additionally the exhaust air ductwork is connected to the radwaste effluent holdup tanks to prevent the potential build up of gaseous radioactivity or hydrogen gas within these tanks. The exhaust fans normally discharge the auxiliary/annex building exhaust air into the plant vent at an approximate flowrate of 16.42 m^3s^{-1} (34,900 cfm).

The supply and exhaust ducts are configured so that each subsystem may be independently isolated. If the radiation monitors in either duct system detect a high level of radiation, the subsystem extract is diverted to the VFS. This allows filtration by both HEPA filters and charcoal filters which provides abatement of both particulate emissions and radioiodine gases. The VAS and VFS systems may also be switched manually if particular operations are being undertaken which could result in release of activity.

In addition to the duct monitors, the following area monitors will also provide a VFS actuation signal to divert the VAS exhaust to the HEPA filters and charcoal filters of the VFS system:

- Primary Sampling Room
- Chemistry Laboratory
- Fuel Handling Area 1
- Auxiliary Building Rail Car Bay/Filter StorageArea
- Liquid and Gaseous Radwaste Area
- Annex Staging and Storage Area
- Fuel Handling Area 2

The purpose of using these area monitors to actuate the switch from VAS to VFS upon contamination detection improves the reliability of the switching system and reduces the duration of potentially untreated atmospheric releases from ~30 seconds to ~15 seconds.

4.3.2.2.3 Health Physics and Hot Machine Shop HVAC System

The Health Physics and Hot Machine Shop HVAC System (VHS) exhaust air system consists of two 100% capacity exhaust fans sized to allow the system to maintain negative pressure. HEPA filtration is not provided on the HVAC system and normally air discharges directly to the plant vent at a flow rate of 6.84 m^3s^{-1} (14,500 cfm). However, in the event that duct monitors or area monitors detect contamination the VHS will be diverted to the VFS to allow filtration by both HEPA filters and charcoal filters.

The hot machine shop provides a location within the controlled area for repair and refurbishment of items of equipment from within the controlled area. The facility has a dedicated decontamination facility which has HEPA filtration and a glovebox which also has HEPA filtration. Individual machine tools have local exhaust ventilation also equipped with HEPA filters with each individual machine operating at an exhaust flow rate from of 0.85 m^3s^{-1} (1800 cfm).

4.3.2.2.4 Radwaste Building Ventilation (VRS)

The Radwaste Building HVAC System (VRS) supplies and exhausts air from the radwaste building. The Radwaste Building has three potential sources of radioactive contamination, these are:

- Tanks for low level liquid effluent for monitoring and sentencing
- Area for loading packaged solid LLW into containers
- Portable or permanently installed equipment for processing LLW

The VRS general extract may contain significant airborne activity either during normal operation or fault conditions if the portable radwaste equipment is not properly operated. Extract air from the building equipment will be by means of low level extract grilles and conveyed through high integrity ductwork to HEPA filters and discharged to the main plant exhaust stack by two 50 percent extract fans. Dedicated HEPA filtered extracted branches will provide extract from the waste sorting cabinets.

4.3.2.2.5 ILW Store Ventilation

The ILW store will be equipped with two HEPA filters in series to remove radioactive particulates present in the ILW building atmosphere.

4.3.2.3 BAT – Ventilation Systems

4.3.2.3.1 BAT – HEPA Filter Selection for Ventilation

The HEPA filter housing design will be capable of holding a range of different specification filters. Higher specification HEPA filters are available than those shown in Table 4-8. However, these filters may increase differential pressure and have shorter replacement intervals than the specified filters. This would result in increased energy use by the extraction fans and larger filter element waste volumes requiring disposal as LLW. The final choice of filter element is best determined by operator experience when the optimum balance between filter performance, cost of filters, and cost of filter disposal can be evaluated.

4.3.2.3.2 BAT – Switching Extraction to VFS Ventilation System upon Detection of Activity

The normal operating condition is one in which radioactivity is not detected within the radiologically controlled areas of the auxiliary and annex buildings. Under these circumstances the air extracted by the VHS and the VAS auxiliary/annex building ventilation subsystem is emitted to atmosphere via the plant vent without treatment. The air extracted from the VAS fuel handling area ventilation subsystem is emitted to atmosphere after passage through HEPA filters. Upon detection of radioactivity all these ventilation systems are switched to pass through the VFS abatement system. This ensures that contaminated air is both filtered and passed through charcoal beds to absorb radioiodine gases. The switching mechanism is triggered by detection of radioactivity in a number of duct and area monitors to ensure reliable and rapid switching (~15 seconds) of the ventilation system to the VFS abatement system.

The advantage of this system is that the exhaust air filtration units of the VFS will only be used to filter air from radiologically controlled areas outside containment upon detection of contamination. This prolongs the life of the filters and charcoal adsorber and minimises the generation of LLW. The approach is consistent with current UK guidance for the design of ventilation systems [Ref. 19]; Clause 2.4.1b of which states:

"for environmental protection (and also cost reasons) it is now accepted policy to minimize radioactive waste arisings as far as practicable; in particular, contaminated HEPA filters, being of low density are very expensive to store or dispose of as radioactive waste"

It is estimated that the alternative of providing the VHS and VAS auxiliary/annex building ventilation subsystem with continuous HEPA filtration would increase radioactive waste arisings by 476m^3 with an associated LLW disposal cost of £1,776,900.

The total cost of adding HEPA filtration to the VAS auxiliary/annex building ventilation subsystem is estimated to be ~£4,968,700 over the 60 year plant operating life. For the VHS the cost is estimated to be ~£1,656,200 on the same basis. These estimates exclude engineering costs, operating and maintenance and decommissioning costs.

It is concluded that the proposed switching of the VHS and VAS ventilation subsystems to the VFS abatement system is the BAT solution.

4.3.3 Liquid Radwaste Treatment Systems

The WLS provides the capability to reduce the amounts of radioactive nuclides released in the liquid wastes through the use of demineralisation and time delay for decay of short-lived nuclides.

The WLS processes the categories of radioactively contaminated wastes as shown in Table 4-9.

The principal process equipment for treating liquid radwaste is a set of four ion exchangers connected in series. There is a waste pre-filter upstream and a waste post-filter downstream of the ion exchangers. The ion exchangers and consist of the following:

- One specific ion exchanger (containing activated charcoal on a zeolite resin) that acts as a deep-bed filter and removes oil from floor drain wastes

- One cation bed ion exchanger

- Two mixed bed ion exchangers

The maximum processing capacity of the ion-exchangers is 408 m^3d^{-1}. This capacity provides an adequate margin for processing a surge in the generation rate of this waste.

Design flexibility exists to manually bypass, under procedural control, any of these ion exchangers, as well as to interchange the order of the last two mixed beds, to provide complete usage of the resin. The WLS piping also permits connection of mobile processing equipment. When liquid wastes are processed by mobile equipment, the treated liquid waste is returned to the WLS for eventual discharge to the environs, or to an ultimate disposal point for liquids that are to be removed from the plant site.

The detergent waste subsystem collects wastes that are generally high in dissolved solids, but low in radioactivity, from plant hot sinks and showers and some cleanup and decontamination processes. The detergent wastes are generally not compatible with the ion exchange resins and are collected in the chemical waste tank (~34 m^3). Normally, these wastes are sampled. If the detergent waste activity is below acceptable limits, the waste can be discharged without processing. When detergent waste activity is above acceptable limits and processing is necessary, the waste water may be transferred to a waste holdup tank and processed in the same manner as other radioactively contaminated waste water, if onsite equipment is suitable to do so. If onsite processing capabilities are not suitable for the composition of the detergent waste, processing can be performed using mobile equipment brought into the radwaste building, or the waste water can be shipped offsite for processing. After processing by the mobile equipment, the water may be transferred to a waste holdup tank for further processing or transferred to a monitor tank for sampling and discharge.

Radioactively contaminated chemical wastes are normally generated at a low rate and collected in the chemical waste tank shared with detergent wastes. Chemicals are added to the tank as needed for pH or other chemical adjustment. The design includes alternatives for processing or discharge. These wastes may be processed onsite, without being combined with other wastes, using mobile equipment. When combined with detergent wastes, they may be treated like detergent wastes, as described above. If onsite processing capabilities are not suitable, processing can be performed using mobile equipment, or the waste water can be shipped offsite for processing.

Process discharge is normally aligned to one of the three monitor tanks. The release of treated liquid waste from any monitor tank to the environment is permitted only when sampling of the subject tank's contents indicates that such a release is permissible.

All WLS releases are monitored by a radiation monitor prior to discharge. The monitor is located on the common discharge line downstream of the WLS to monitor tank limits for radionuclide concentrations in liquid effluents discharged into unrestricted areas. These radiation monitors will provide a signal to terminate liquid radwaste releases if the discharge concentration in the line exceeds a predetermined set point.

Effluent meeting the local discharge requirements is pumped from the monitor tanks in a controlled fashion to the cooling water return from the circulating water system (CWS). The monitor tank pumps have a design flow rate of ~545m^3 per day, although the average daily liquid radioactive waste release rate is ~8m^3 per day. The once through cooling water flow rate is 600,000 U.S. gallons per minute (136,275m^3/h). It follows that the cooling water stream provides a substantial dilution of the discharged effluent before release to the environment.

4.3.3.1 BAT Assessment for Liquid Radwaste Treatment

4.3.3.1.1 Ion Exchange vs. Evaporation

A comparison of typical flow sheets for evaporation and ion exchange is shown in Figure 4-2. The relative merits of ion exchange and evaporation has been evaluated by Westinghouse and the results are reported in Table 4-10.

The standard AP1000 NPP design does not have evaporators based on considerations shown in Table 4-10 and because it contradicts the AP1000 NPP overriding principle of safety and simplicity. Compared to traditional evaporator-based liquid radwaste system, the ion-exchange based AP1000 NPP system provides effectiveness and simplicity, and will tend to minimise operator doses and solid radwaste arisings. The complexity of the traditional evaporator design leads to significant maintenance with associated occupational radiation exposure, and also gives more opportunity for operator errors. The relatively passive nature of the ion exchange-based AP1000 NPP system provides effective operation without the issues of the evaporator-based system and at lower capital and operating cost.

The fact that the generic site for the AP1000 NPP is a coastal site and not a river site also lessens the value of using evaporators for minimising the discharge of boric acid. Unlike river water, seawater already contains significant boron concentrations.

At Sizewell B two evaporators were constructed: one for recycling boric acid from the RCS, and one for abatement of liquid radioactive waste. Evaporation of liquid for either purpose is not currently considered BPM or ALARP, and the evaporators are not in use. This is because the benefit of reducing liquid discharges, in terms of the consequent small reduction of public dose, is much less than the potential harm of increased operator doses. In addition, the small reduction in public dose would not justify the cost of processing (evaporator and encapsulation) and the cost of providing sufficient high quality steam to run the evaporators.

The ion exchange treatment process has been shown to effectively control off-site discharges. For the generic site, it has been demonstrated that the AP1000 NPP effluent discharges can be released to the coastal environment without contributing excessively to marine ecosystem dose rates [Reference 7].

It is concluded that the proposed WLS treatment system using ion exchange beds and filtration rather than evaporation is BAT.

4.3.3.1.2 Enriched Boric Acid vs Natural Boric Acid

The AP1000 NPP is designed not to require a high quality boron source. Natural boric acid is used rather than very costly B-10 enriched boric acid.

The use of B-10 enriched boric acid has the potential of reducing the concentration of boron required as a moderator in the RCS. Enriched boric acid typically contains 60% B-10 compared to 20% B-10 in standard boric acid. As B-10 is the effective reactor moderator, the use of enriched boric acid has the potential for reducing the boron concentration in the RCS, at maximum, by a factor of three.

In principle, the use of enriched boric acid also reduces the amount of lithium hydroxide required for pH control by a factor of three. Reduced lithium hydroxide concentration would reduce the potential for tritium formation associated with neutron absorption by lithium isotopes (see Section 4.3.1.5 and Appendix A.1). However, the AP1000 NPP employs other more effective measures to minimise tritium formation including:

- Use of gray rods for mechanical shim control which reduces the quantity of boric acid required for chemical shim control (see Sections 4.1.3.3 and 4.3.1.6).

- Use of Li7OH rather than natural lithium hydroxide for pH control. This substantially reduces potential for tritium formation from neutron absorption by Li-6 present in natural lithium hydroxide (see Section 4.3.1.5 and Appendix A.1).

The cost of enriched boric acid is more than two hundred times the cost of natural boric acid. Since lithium hydroxide is a strong base and boric acid is a weak acid, only a small quantity of lithium hydroxide is needed to adjust the pH of boric acid concentrations in the RCS. It is more cost effective to use slightly more Li7OH for the pH control of natural boric acid than it is to incur the high cost of enriched boric acid with lower Li7OH use.

4.3.3.1.3 Boron Discharge vs. Boron Recycle

The requirement for a reduction in the use of boron has been driven by U.S. users who see a capital and operating cost benefit in the reduced use of boron, as well as a major reduction in the complexity of the plant.

The AP1000 NPP adopts several approaches which minimise the production of liquid radwaste before the treatment by the WLS (see Section 4.1). In particular, the use of mechanical shim control rather than chemical shim control during normal load follow operations substantially reduces the quantities of boron use as a moderator. This reduces the amount of boron that needs to be removed from the reactor coolant water and, therefore, reduces the amount of liquid radioactive waste produced.

The AP1000 NPP is designed not to require a high quality boron source. Natural boron is used rather than costly boron enriched in the B-10 isotope. This reduces the economic incentive for recycling boron.

Boron recycling requires a significant amount of additional equipment. The borated water cannot be reused until the start of the next fuel cycle and must be stored for long periods. This storage presents an additional safety issue and an additional source of operator dose which is not considered ALARP. The additional equipment also presents increased operator dose during maintenance and decommissioning.

Assuming the monitor tanks contain water with the upper limit of 2700mg/l of boron and the effluent is discharged at 22.7m^3/h into in the seawater cooling return flow of 136,275m^3/h, the boron concentration in the cooling return would be increased by 450 µg/l. At an average liquid radwaste effluent flow rate of 8m^3/d, such as discharge would only occur for 128 hours per year. It is concluded that the boron discharge is negligible in relation to the annual average environmental quality standard of 7000µg/l for the protection of saltwater life [Reference 16], and that discharge of boron to seawater meets BAT and ALARP criteria.

4.3.3.1.4 Cartridge Filtration vs Cross Flow Filtration

The WLS incorporates an after filter downstream of the ion exchangers to collect particulate matter, such as resin fines. The disposable filter cartridges have a design filtration efficiency of 98 percent removal of 0.5µm particles. The radioactive particulate load in the WLS influent is already reduced by passage through the pre-filter, deep bed filter, and three ion exchange beds before the after filter. The use of cartridge filters offers a low pressure system that is suitable for the low flow rates (~8m^3/day) associated with the WLS. The filters are readily replaceable and treated as LLW.

Cross-flow filtration techniques of microfiltration, ultrafiltration, nanofiltration, and reverse osmosis potentially offer increasingly effective particulate removal efficiency (ranging from 0.1 µm to <0.001 µm) compared to cartridge filtration. All these techniques use membrane processes that segregate a liquid that permeates through a membrane from a concentrate which is retained. The driving force of the process is the pressure difference across the membrane. The disadvantages of these processes are as follows:

- High pressure systems to drive the filtration process which carries with it increased potential for leaks. The pressure requirements increase as follows: microfiltration < ultrafiltration < nanofiltration < reverse osmosis.

- Complicated return, recycling and bleed system designs to deal with the concentrate stream

- Polymeric membranes used, particularly in ultrafiltration, nanofiltration and reverse osmosis, are subject to degradation by decay of captured radioactive particulates.

- The complexity of these systems relative to the proposed cartridge filtration system has the potential for greater levels of maintenance and higher associated operator dose.

- More equipment that will become radioactive waste during decommissioning

- Higher capital and operating costs than cartridge filtration

It is concluded that the proposed use of cartridge filters is BAT for filtration after the ion exchange beds.

4.3.4 Solid Radwaste Treatment Systems

4.3.4.1 LLW and ILW

The solid waste treatment systems for LLW and ILW are based on the techniques identified as part of the BAT assessment described in Section 6 of this report. The "UK AP1000 Environment Report" [Reference 5, subsection 3.5.7] provides more detail on the treatment systems for LLW and ILW.

4.3.4.2 HLW

The Government's policy on reprocessing is that the decision to reprocess or hold the spent fuel in long-term storage is a matter for the commercial judgment of the owner of the spent fuel subject to meeting the necessary regulatory requirements [Reference 17].

There is no intention to reprocess spent fuel from the AP1000 NPP. It is planned that the operators will safely store this fuel at their reactor sites until a permanent disposal repository for spent nuclear fuel is built. This allows flexibility by allowing the decision to reprocess or permanently dispose of the HLW to be deferred and reassessed when the options become clear in the future.

After spent fuel is removed from the reactor, it will be stored in the fuel storage pool for a period of ~18 years. This provides adequate time for the proposals set out below to be reviewed and amended according to conditions prevailing at the time a decision is required.

A facility for the storage of spent fuel for the operational period of the plant and beyond is being designed because spent fuel is not expected to be reprocessed. The key BAT decisions for the spent fuel storage facility is whether to store the fuel wet or dry and whether to store the fuel above or below ground as follows:

- Although fuel transfers are all carried out underwater, it is preferred to store fuel dry in canisters and under an inert gas atmosphere for the long-term storage of the fuel. This avoids the corrosion issues associated with long-term wet storage.

- Underground storage is the preferred choice for long-term storage because it has the advantage of providing greater levels of shielding and a more secure solution with respect to aircraft impact and other catastrophic events. However, the disadvantages of underground storage relating to control of groundwater and flood risk may become important in the decision process depending upon the site. These issues need to be considered carefully at the site-specific design stage.

For the GDA, the spent fuel system proposed for the generic site is a dry, underground storage system and comprises:

- Flask loading equipment within the AP1000 NPP.
- Suitable flask transportation vehicles and equipment.
- A seismically qualified below ground storage facility.

Westinghouse is proposing the dry spent fuel storage system from Holtec with long-term storage inside an underground cylindrical cavity [Reference 18]. However, the AP1000 NPP operator may choose another option. The flask handling equipment within the AP1000 NPP can accommodate a variety of flask types.

4.4 Comparison with Existing Plant

4.4.1 Gaseous Discharges

The gaseous discharges from the AP1000 NPP were compared with those from the following operating plants (Table 4-11):

- Sizewell B
- South Texas 1
- Braidwood 1
- Cook 1
- Vogtle 1

These plants were selected for comparison to the generic AP1000 NPP because Sizewell B is a pressurised water reactor in the United Kingdom and South Texas 1, Braidwood 1, Cook 1, and Vogtle 1 are more recently built Westinghouse PWRs in the U.S. When the values are normalized to an annual basis and 1000 MW output, the AP1000 NPP has lower discharges than all but one of the plants, and essentially the same as the remaining one.

4.4.2 Liquid Discharges

The liquid discharges from the AP1000 NPP were compared with those from the following operating plants (Table 4-12):

- Sizewell B
- South Texas 1
- Braidwood 1
- Cook 1
- Vogtle 1

The predicted liquid discharges from the AP1000 NPP are compared in Tables 4-13 and Table 4-14 with published discharges from European nuclear reactors operating over the period 1995-1998 [Reference 10]. The tritium data in Table 4-13 indicates that the predicted AP1000 NPP discharges are similar to Sizewell B discharges, but above the European average for all European PWRs. The predicted AP1000 NPP tritium discharges are less than the Magnox and advanced gas reactors (AGRs), but higher than discharges from BWRs. It is practically very difficult to reduce discharges of tritium. The radiological impact of tritium is relatively small and radiological impact of discharges is usually very low.

In Europe many PWRs are located on major rivers and not on coastal sites.. It is common for these reactors to be equipped with evaporators to minimise radioactive liquid and boric acid discharges.

Table 4-14 compares the predicted non-tritium radioactive liquid discharges from the AP1000 NPP against published data for European nuclear power stations between 1995 and 1998 [Reference 10]. The results indicate that the AP1000 NPP emissions are predicted to be approximately 50 percent of the average PWR discharges. The predicted discharges are also considerable lower than the average Magnox, AGR, BWR and Sizewell B discharges.

The liquid discharges from the AP1000 NPP are compared with published discharges from European PWRs of isotopes other than tritium in Figure 4-3 [Reference 10]. The results indicate that even without an evaporator, the predicted non-tritium liquid discharges from the AP1000 NPP (2.5GBq/GWa) compare favorably with PWRs with the lowest reported releases between 1995 and 1998.

4.4.3 Comparison of Solid Waste with Sizewell B

Sizewell B is the only PWR in operation in the United Kingdom. The operators, British Energy Generation Ltd. (BEGL), carried out a review of the control and impact of the discharge and disposal of radioactive waste at Sizewell B in 2005 [Reference 8]. The review was prepared as a submission of information to the Environment Agency to enable their review of Radioactive Substances Act 1993 authorisations. In 2006, the Environment Agency published their decision document and authorisations regarding future regulation of disposals of radioactive waste at UK nuclear power stations [Reference 9]. This review commented on the BPEO and BPM proposed by BEGL for control of emissions and discharges of radioactive wastes from Sizewell B.

Table 4-15 presents the BPEO issues identified for gaseous and liquid wastes at Sizewell B and compares them with the practices proposed for the AP1000 NPP. The table also provides a summary of the Environment Agency comments on the Sizewell B BPEO issues.

Table 4-1

BUILDING VENTILATION SYSTEMS

Area	Ventilation System	Ventilation Abatement Provisions
Nuclear Island Nonradioactive	VBS	No filtration.
Annex/Auxiliary Building Nonradioactive	VXS	No filtration
Diesel Generator Building	VZS	No filtration
Containment	VCS	No discharge outside containment.
Containment	VFS	High efficiency and HEPA filtration
Health Physics and Hot Machine Shop – Gloveboxes	VHS	HEPA filtration
Health Physics and Hot Machine Shop – Machine tools	VHS	HEPA filtration
Health Physics and Hot Machine Shop – Remaining space	VHS	No filtration but diversion to HEPA and charcoal filtered standby system (VFS) on High radiation signal.
Radwaste Building	VRS	HEPA filtration
Turbine Building – Bay 1 area	VTS	No filtration
Turbine Building – Remaining space	VTS	No filtration
Auxiliary/Annex Building Radiologically Controlled Area – Fuel handling area	VAS	HEPA filtration and diversion to HEPA and charcoal filtered standby system (VFS) on High radiation signal.
Auxiliary/Annex Building Radiologically Controlled Area – Remaining space	VAS	No filtration but diversion to HEPA and charcoal filtered standby system (VFS) on High radiation signal.

Table 4-2

GASEOUS RADIATION EMISSIONS OVER THE AP1000 NPP FUEL CYCLE

Sum Of Monthly Gaseous Radiation Breakdown(1)

	Total		
Month	**RCS (GBq)**	**NON-RCS (GBq)**	**Total (GBq)**
0	-----	-----	-----
1	121	448	568
2	128	448	575
3	136	448	583
4	145	448	592
5	155	448	602
6	167	448	614
7	180	448	628
8	197	448	644
9	216	448	664
10	240	448	687
11	269	448	717
12	307	448	755
13	357	448	805
14	427	448	875
15	532	448	980
16	704	448	1152
17	1046	448	1494
18	2079	448	2527
Total	7405	8059	15463

Note:

1. Data from Table 3.3-10 of "UK AP1000 Environment Report" [Reference 5]

Table 4-3

NON-RADIOACTIVE WATER SYSTEMS

Water System	System Abbreviation	Comment
Service water system	SWS	For the generic coastal site, the SWS supplies once through seawater cooling[(1)] to remove heat from the non-safety-related component cooling water system heat exchangers in the turbine building.
Component cooling water system	CCS	The CCS is a nonsafety-related closed loop cooling system that transfers heat from various plant components to the service water system during normal phases of operation. The CCS also provides a barrier against leakage of service water into primary containment and reactor systems. Leakage of reactor coolant into the CCS is detected by a radiation monitor on the common pump suction header, by routine sampling, or by high level in the surge tank. Excessive leakage from the CCS causes the water level in the component cooling water surge tank to drop and a low level alarm to be actuated. Makeup water is added automatically to the component cooling water system as required.
Demineralised water treatment system	DTS	The DTS receives water from the raw water system, processes this water to remove ionic impurities, and provides demineralised water to the demineralised water transfer and storage system. The treatment system comprises cartridge filters, reverse osmosis units and electrodeionisation systems. The reject flow or brine from the first reverse osmosis unit is discharged to the waste water system.
Potable water system	PWS	The PWS is designed to furnish water for domestic use and human consumption. No interconnections exist between the potable water system and any potentially radioactive system or any system using water for purposes other than domestic water service. Discharges occur from end use drains or system leakage.

Note:

1. The use of cooling towers may be considered at locations where an alternative water source for the raw water system is available. This may be considered at the site specific design stage for suitable sites.

Table 4-3 (cont.)

NON-RADIOACTIVE WATER SYSTEMS

Water System	System Abbreviation	Comment
Sanitary drainage system	SDS	The SDS is designed to collect the site sanitary waste for treatment, dilution and discharge. The sanitary drainage system does not service facilities in radiologically controlled areas. The sanitary drainage system transports sanitary waste to the waste treatment plant, the waste treatment plant is site specific and is outside the scope of the generic site AP1000 NPP application.
Central chilled water system	VWS	The VWS supplies chilled water to the HVAC systems and is functional during reactor full-power and shutdown operation. The system consists of two closed loop subsystems: a high cooling capacity subsystem, and a low cooling capacity subsystem. Discharges occur as a result of blowdown to maintain water chemistry or leakage.
Turbine building closed cooling system	TCS	The TCS is a closed loop system which provides chemically treated, demineralised cooling water for the removal of heat from non-safety related heat exchangers in the turbine building and rejects the heat to the circulating water system. The cooling water is treated with a corrosion inhibitor and uses demineralised water for makeup. Discharges occur as a result of blowdown to maintain water chemistry or leakage.
Waste water system	WWS	The waste water system collects and processes equipment and floor drains from non-radioactive building areas. It is capable of handling the anticipated flow of waste water during normal plant operation and during plant outages. Effluent is collected in the turbine building sumps. The sumps are discharged via an oil separator. The waste oil is collected in a temporary storage tank before trucks remove the waste for offsite disposal. The waste water from the oil separator flows by gravity to a waste water retention basin for settling of suspended solids and treatment before discharge, if required. The effluent in the retention basin is pumped to the plant cooling water outfall. In the event radioactivity is detected in the discharge from the sumps; the waste water is diverted from the sumps to the WLS for processing and disposal.

Table 4-3 (cont.)

NON-RADIOACTIVE WATER SYSTEMS

Water System	System Abbreviation	Comment
Hot water heating system	VYS	The VYS supplies heated water to selected non-safety-related air handling units and unit heaters in the plant during cold weather operation and to the containment recirculating fans coil units during cold weather plant outages. Level instrumentation on the surge tank provides both high and low-level alarms. At tank low-level, makeup is provided from the demineralised transfer and storage system.
Circulating water system	CWS	The CWS is a once through seawater cooling water system with supply to the main condenser to condense the steam exhausted from the main turbine. The cooling water system is a site-specific design.
Steam generator blowdown system	BDS	The BDS assists in maintaining acceptable secondary coolant water chemistry during normal operation and during anticipated operational occurrences of main condenser in-leakage or primary to secondary steam generator tube leakage. If significant radioactivity is detected in secondary side systems, blowdown is redirected to the WLS. However, normal operation is for the blowdown from each steam generator to be processed by a regenerative heat exchanger to provide cooling and an electrodeionization demineralising unit to remove impurities from the blowdown flow. The blowdown fluid is then normally recovered for reuse in the condensate system. Blowdown with high levels of impurities can be discharged directly to the WWS. A small waste stream from the electrodeionization system may also be directed to the WWS or the WLS.

Table 4-4

LIQUID RADIATION DISCHARGES OVER THE AP1000 NPP FUEL CYCLE

Month	Sum of Monthly Liquid Radiation Breakdown(1)		
	RCS (GBq)	NON-RCS (GBq)	TOTAL (GBq)
0	-----	-----	-----
1	128	2345	2473
2	135	2345	2481
3	144	2345	2489
4	153	2345	2499
5	164	2345	2510
6	177	2345	2522
7	191	2345	2537
8	208	2345	2554
9	229	2345	2574
10	254	2345	2600
11	285	2345	2631
12	326	2345	2671
13	379	2345	2724
14	453	2345	2799
15	564	2345	2909
16	747	2345	3092
17	1109	2345	3455
18	2205	2345	4550
Total	7852	42217	50070

Note:

1. Data from Table 3.4-7 of "UK AP1000 Environment Report" [Reference 5]

Table 4-5

NON-RADIOACTIVE EFFLUENT DISCHARGES

System	Non-Radioactive Effluent Description	Physical/Chemical Description	Estimated Quantity (m^3/h)	
			Normal	Maximum
WWS	Condensate demineraliser rinses and backwashes	Demineralised water with minor solids	0.01	466
BDS	Steam generator blowdown	Secondary side coolant(1)	4.2	42
BDS	Condensate demineraliser startup bypass flow	Off-specification demineralised water	26	82
DTS	Reverse osmosis (RO) and electrodeionization reject	Off-specification demineralised water	13	41
BDS	Fire testing drains	Demineralised water with minor solids	0.1	170
multiple	Turbine island waste water	Demineralised water with minor solids	18	74
CDS	Condenser water box drain	Demineralised water with minor solids	0	250
SWS	Strainer backwash	Demineralised water with minor solids	1.0	681
CWS	Strainer backwash	Demineralised water with minor solids	2.2	413
CPS	Condensate polisher rinse	Demineralised water with minor solids	0.01	466
SWS	Service water system	Non-contact once through seawater cooling	2385	4770
CWS	Circulating water system	Non-contact once through seawater cooling	136,275	136,275

Note:

1. Normally non-radioactive – diverted to liquid radwaste treatment if contamination detected

Table 4-6

SUMMARY OF MAIN SOLID RADIOACTIVE WASTE PRODUCED BY THE AP1000 NPP

Description of Waste	Radioactive Waste Classification	Frequency	Normal Volume per Unit Frequency (m^3)	Maximum Volume per Unit Frequency (m^3)	Volume per Life of Plant (m^3)
Spent fuel rods	HLW	40%/18 months	13.7		549
Ion exchange resin	ILW	Annual	7.8	15.6	561
Gray rod cluster	ILW	Once/20 yr	1.7		5.1
Control Rod Cluster	ILW	Once/20 yr	5.6		16.9
Wet granular carbon	ILW	Annual	0.6	1.1	41
Filter cartridge – metallic cylinder	ILW	Annual	0.2	0.4	13.7
Compactable paper, tape, clothing, plastic, elastomers	LLW	Annual	135	206	8924
Non-compactable metallic items, glass, wood	LLW	Annual	6.6	10.6	455
HVAC filter – uncompacted fibreglass/metal	LLW	Various			761
Condensate Polisher spent resin	LLW	Annual	3.9	7.7	69.3
Dry granular carbon	LLW	Annual	0.3	3.3	54.3
HVAC filter – granulated charcoal	LLW	Once/10 yr	4.9		29.1
Compressible rigid plastic – gaskets, valve packing, insulation	LLW	Various			7.6
Electrodeionisation Unit – resin/membrane module	LLW	Once/12 yr	1.7		10.8
Heat exchanger insulation	LLW	Once/60 yr	8.4		8.4

Table 4-6 (cont.)

SUMMARY OF MAIN SOLID RADIOACTIVE WASTE PRODUCED BY THE AP1000 NPP

Description of Waste	**Radioactive Waste Classification**	**Frequency**	**Normal Volume per Unit Frequency (m^3)**	**Maximum Volume per Unit Frequency (m^3)**	**Volume per Life of Plant (m^3)**
Filter – pleated polyester	LLW	Annual	0.1		5
Wet granular particles – sludge	LLW	Annual	0.03	0.1	2.4
Waste Oil	LLW	Once/5 y	0.15		1.8
Mechanical pump seal – SiC	LLW	Once/5 yr to Once/30 yr	0.05		0.58
Pump diaphragms – Buna n	LLW	Once/5 yr	0.04		0.47
Degasifier Separator – canned pump	LLW	Once/60 yr	0.06		0.06
Resin transfer screw pump	LLW	Once/10 yr	0.003		0.02

Table 4.3-1

SUMMARY OF MAIN SOLID NON-RADIOACTIVE WASTE PRODUCED BY THE AP1000 NPP

Description of Waste Radioactive Waste Classification	**Frequency**	**Normal Volume per Unit Frequency (m^3)**	**Volume per Life of Plant (m^3)**
HVAC filters (fibreglass/metal)	various	various	5209
Battery (lead acid)	Once/20 y	324	630
Lube oil	Once/25 y	79.5	159
Reverse osmosis modules	Once/7 y	15.77	135.2
Electrodeionisation/reverse osmosis filter cartridges	Once/6 months	0.39	45.65
HVAC filters (charcoal)	Once/10y	4.86	29.12
Valve Packing – compressible rigid plastic	Once/5 y	1.14	13.7
Electrodeionisation (resin/membrane module)	Once/12 y	1.34	6.68
Door/hatch gaskets (fibreglass cloth)	Once/60 y	1.16	1.16
Main feedwater pump seals (silicon carbide)	Once/5 y	0.056	0.68
Heat Exchanger gaskets (neoprene)	Once/10 y	0.062	0.37

Table 4-8

SPECIFICATION OF CONTAINMENT FILTRATION SYSTEM ELEMENTS

	Pre-high Efficiency Filter	**HEPA Filter**	**Charcoal Filter**	**Post-high Efficiency Filters**
Design type	High efficiency	HEPA	Type III rechargeable cell	High efficiency
Design code or standard	ASME N509	ASME N509	ASME N509	ASME N509
Dimensions (Approximate maximum for each unit)	10.7m x 2.0m x 1.7m (35′ x 6.5′ x 5.6′)			
Construction material/ filter material	Utility specific	Utility specific	Utility specific	Utility specific
Filter pass (pore) size	Utility specific	Utility specific	Utility specific	Utility specific
Typical flowrate per unit (m^3/h)	6800	6800	6800	6800
Efficiency	80% minimum ASHRAE efficiency	>99.97% 0.3μm DOP	90% Decontamination efficiency	95% 0.3μm DOP
Monitoring of efficiency	Periodic DOP testing	Periodic DOP testing	Periodic DOP testing	Periodic DOP testing
Detection of filter blinding	Differential pressure instrument	Differential pressure instrument	Radiation monitoring in the plant vent	Differential pressure instrument
Typical 'in-service' periods	Once a week for 20 hours			
Arrangement to take filter out of service	Both filter units are 100% redundant. When one is being maintained, it can be bypassed, and the other can be used.			

Table 4-9

RADIOACTIVELY CONTAMINATED WASTES PROCESSED BY LIQUID RADWASTE SYSTEM

Source	Expected Input Rate (m^3/d)	Receiving Tank	Comment
CVS letdown	1.65	Effluent holdup tanks (3 x $106m^3$)	Borated waste water from the RCS effluents released through the CVS
Leakage inside containment (to reactor coolant drain tank)	0.04		
Leakage outside containment	0.3		
Sampling drains	0.8		Primary sampling system sink drain
Reactor containment cooling	1.9	Waste holdup tanks (2 x $57m^3$)	
Spent fuel pool liner leakage	0.09		
Miscellaneous drains	2.6		Floor drains from various building sumps and equipment drains
Detergent waste	0.9	Chemical waste tank (1 x $34m^3$)	From hot sinks and showers, and some cleanup and decontamination processes
Chemical wastes	0.008		From the laboratory and other relatively small volume sources
Contaminated secondary system effluent	0 (normal condition)	Waste holdup tank	The WLS does not normally process non-radioactive secondary system effluent. The SG blowdown system and the turbine building drain system normally handle secondary system effluents. However, radioactivity can enter the secondary systems from steam generator tube leakage. If significant radioactivity is detected in secondary side systems, blowdown is redirected to the WLS for processing and monitored disposal.

Table 4-10

BAT COMPARISON OF EVAPORATORS AND ION EXCHANGE FOR THE TREATMENT OF LIQUID RADWASTE

	Natural Circulation Evaporators	**Forced Circulation Evaporators**	**Ion Exchange**
Where applied for radwaste processing	Traditionally applied in U.S. PWRs – later replaced in some with ion exchange or forced circulation evaporators	Japan; occasionally in U.S.	Newer U.S. plants.
Processing	Evaporator removes all solids in waste stream. Concentrates to 12wt% "slurry" which is drummed or solidified	Evaporator removes all solids in waste stream. Concentrates to 12wt% "slurry" (or higher) which is drummed or solidified	Ion exchange process removes activity from fluid. Non-specified solids (for example, concrete dust, sand) and boric acid pass through to discharge
Effectiveness	Acceptable decontamination factor 100-500	Good decontamination factor 100-1000	Good to excellent with appropriate usage. Decontamination factor 100 – 400 for single vessel, higher for multiple vessels in series
Flexibility	Poor – many inputs can upset evaporator (for example, detergents, oil)	Excellent – same process for all inputs	Excellent, but requires intelligent control: • Oils must be segregated – will ruin resin • Most detergents must be segregated • Most effective use comes through "tuning" selected resins for prevailing conditions
Capital Cost	High – typically provided as custom-built, skid-mounted units	Very high – custom design and construction; essentially a complex system unto itself	Low – ion exchange vessels only Cost ~50% evaporator
Operating Cost	Moderate – steam/energy	Moderate – steam/energy	High – resins Low – equipment – much less equipment and less active equipment to maintain

Table 4-10 (cont.)

BAT COMPARISON OF EVAPORATORS AND ION EXCHANGE FOR THE TREATMENT OF LIQUID RADWASTE

	Natural Circulation Evaporators	**Forced Circulation Evaporators**	**Ion Exchange**
Safety	Excellent	Excellent	Excellent
Reliability	Poor – 12wt% boric acid operation leads to frequent problems	Good	Excellent
Operability	Poor – problems with foaming and solidification	Good	More complex – Operator should sample holdup tank contents and select ion exchange resins accordingly Higher throughput possible, reducing potential impact on plant availability
Maintainability	Very poor – highly radioactive, no room to work	Moderate – many components, but adequate space is provided	Excellent – only normal maintenance is resin flushing which is remote
Occupational radiation exposure	High	Moderate	Very low
Layout impact	Low – small skid mounted system	Very large – sometimes an entire dedicated building	Low – 4 exchange vessels, 2 filters
Solid radwaste production	High	High (may be lower depending on concentration	Low
Estimated waste volumes for 900MWe Plant	Resins 6 m^3/y Filter cartridges 0.5 m^3/y Evaporator bottoms 102 m^3/y Chemical wastes 1 m^3/y Total 109.5 m^3/y		9 m^3/y 1 m^3/y 0 m^3/y 1 m^3/y 11 m^3/y
Decommissioning	Moderate – complex dismantling of highly radioactive equipment	High – complex dismantling of large highly radioactive equipment	Low – simple decontamination and dismantling of resin tanks
Tritium	Increased transfer of tritium from water to air. Impact of tritium dose more significant in air than water		Greater proportion of tritium in water than air

Table 4-10 (cont.)

BAT COMPARISON OF EVAPORATORS AND ION EXCHANGE FOR THE TREATMENT OF LIQUID RADWASTE

	Natural Circulation Evaporators	**Forced Circulation Evaporators**	**Ion Exchange**
Licensable	Traditionally licensable, but not allowed by U.S. utility requirements document	Acceptable	Acceptable – licensed in U.S. and supported by U.S. utility requirements document Boric acid discharge – must be considered, but probably not an issue for seawater site or enriched boric acid

Table 4-11

COMPARISON OF AP1000 NPP GASEOUS RADIOACTIVE DISCHARGES WITH OTHER NUCLEAR POWER PLANTS

		AP1000 NPP	**Sizewell B**	**South Texas 1**	**Braidwood 1**	**Cook 1**	**Vogtle 1**
Total Discharges	curies	418 /18 mths	2251 /y	260 /y	18 /y	352 /y	69 /y
	GBq	15466 /18 mths	83300 /y	9609 /y	667 /y	13025 /y	2558 /y
Total Discharges Scaled to 1000 MWe and 1 yr	curies	279 /y	1894 /y	208 /y	15 /y	340 /y	59 /y
	GBq	10311 /y	70115 /y	7692 /y	561 /y	12571 /y	2184 /y

Table 4-12

COMPARISON OF AP1000 NPP LIQUID RADIOACTIVE DISCHARGES WITH OTHER NUCLEAR POWER PLANTS

		AP1000 NPP	**Sizewell B**	**South Texas 1**	**Braidwood 1**	**Cook 1**	**Vogtle 1**
Total Discharges	curies	1353 /18 mths	1622 /y	1566 /y	1575 /y	1246 /y	1278 /y
	GBq	50061 /18 mths	60000 /y	57949 /y	58278 /y	46103 /y	47289 /y
Total Discharges Scaled to 1000 MWe and 1 year	curies	902 /y	1365 /y	1252 /y	1327 /y	1203 /y	1093 /y
	GBq	33374 /y	50503 /y	46331 /y	49094 /y	44500 /y	40450 /y

Table 4-13

COMPARISON OF AP1000 NPP LIQUID RADIOACTIVE DISCHARGES OF TRITIUM WITH EUROPEAN NUCLEAR POWER PLANTS BETWEEN 1995 AND1998

	Unit	AP1000 NPP	Sizewell B	All PWR	All Magnox & AGR	All BWR
No. Plants		0	1	73	30	10
Minimum	TBq/GWa	30.5	17.9	0.02	88	0.34
Average	TBq/GWa	33.4	36.1	16.2	357.15	0.83
Maximum	TBq/GWa	35.1	45.9	45.9	463	1.92

Table 4-14

COMPARISON OF AP1000 NPP LIQUID RADIOACTIVE DISCHARGES OF RADIONUCLIDES OTHER THAN TRITIUM WITH EUROPEAN NUCLEAR POWER PLANTS BETWEEN 1995 AND1998

	Unit	AP1000 NPP	Sizewell B	All PWR	All Magnox & AGR	All BWR
No. Plants		0	1	73	30	10
Minimum	GBq/GWa	1.1	16	0	2	0
Average	GBq/GWa	2.4	21.8	4.9	12.2	65.5
Maximum	GBq/GWa	3.5	28	61	28	599

<u>Note:</u>

Data in Tables 4-13 and 4-14 for other nuclear plants extracted from measured data reported in “Implementation of PARCOM Recommendation 91/4 on Radioactive Discharges”, OSPAR Commission 2003 [Reference 10]. AP1000 NPP data is based upon estimated monthly discharge calculations.

Table 4-15

COMPARISON OF BPEO APPROACH AT SIZEWELL B WITH AP1000 NPP BAT APPROACH

BPEO Issue [Reference 8]	Options Considered in Sizewell B BPEO Study [Reference 8]	Sizewell B Recommendation following BPEO Study [Reference 8]	Environment Agency Comment [Reference 9]	AP1000 NPP Comparison
Waste oils	On-site incineration Storage on-site Chemical Oxidation* Biological Oxidation* Pyrolysis*	On-site incineration	Authorised both on-site and off-site disposal, but improvement condition to re-evaluate BPEO attached to use of on-site incinerator. Concern over high water content in PWR waste oils.	Commercial (off-site) incinerator (LLW) or Commercial (off-site) recycling (non-radioactive)
Organic liquid solvents	On-site incineration Storage on-site Biological Oxidation* Pyrolysis*	Storage on-site		Waste stream not forecast, but disposal route could be: Commercial (off-site) incinerator (LLW) or Commercial (off-site) recycling (non-radioactive)
Combustible solid LLW	On-site incinerator Commercial (off-site) incinerator British Energy off-site Incinerator No Incineration; Disposal to the low level waste repository (LLWR)	On-site incineration	Do not consider that BEGL have demonstrated that the continued use of the Sizewell B incinerator is BPEO. Authorised transfer of some incinerable solid waste to a specialist incinerator.	Commercial (off-site) incinerator or No Incineration; disposal to LLWR
Trash solid LLW	Volume reduction and disposal to the LLWR Recycling Decontamination	Disposal to the LLWR	Disposal to the LLWR with volume reduction as a matter for BPM consideration	Disposal to the LLWR

Table 4-15 (cont.)

COMPARISON OF BPEO APPROACH AT SIZEWELL B WITH AP1000 NPP BAT APPROACH

BPEO Issue [Reference 8]	Options Considered in Sizewell B BPEO Study [Reference 8]	Sizewell B Recommendation following BPEO Study [Reference 8]	Environment Agency Comment [Reference 9]	AP1000 NPP Comparison
LLW resins	Encapsulation in cement, disposal to the LLWR	Encapsulation in cement, disposal to the LLWR	Considered encapsulation in cement, disposal to the LLWR to be suitable	Accumulation until end of generation cycle Commercial (off-site) incinerator or encapsulation; storage on site; disposal to LLWR
LLW sludge - oily	Disposal to the LLWR	Subject to batch-specific BPEO	Agreed improvement condition that required BPEO to be submitted before any treatment of LLWR waste, including oily sludge	
LLW sludge - non-oily	Off-site incineration, solids to landfill	Off-site incineration	Off-site incineration or drying and disposal to LLWR authorised	
LLW filters	Disposal to the LLWR, following pre-treatment Incineration	Disposal to the LLWR, following pretreatment subject to BPM constraints	Agreed disposal to the LLWR, following pretreatment , if necessary	Compaction; disposal to the LLWR
Wet ILW	Early encapsulation, long-term storage, disposal to national repository Pyrolysis(1) Acid Digestion(1)	Early encapsulation, long-term storage, disposal to national repository	Agreed long-term storage and no disposal of wet or solid ILW	Early encapsulation; long-term storage; disposal to national repository
Solid ILW	Accumulation until end of generation, encapsulation, storage on site, disposal to national repository	Accumulation until end of generation, encapsulation, storage on site, disposal to national repository		Accumulation until end of generation cycle; encapsulation; storage on site; disposal to national repository

Note:

1. Novel applications with limited relevant information on which to base an assessment

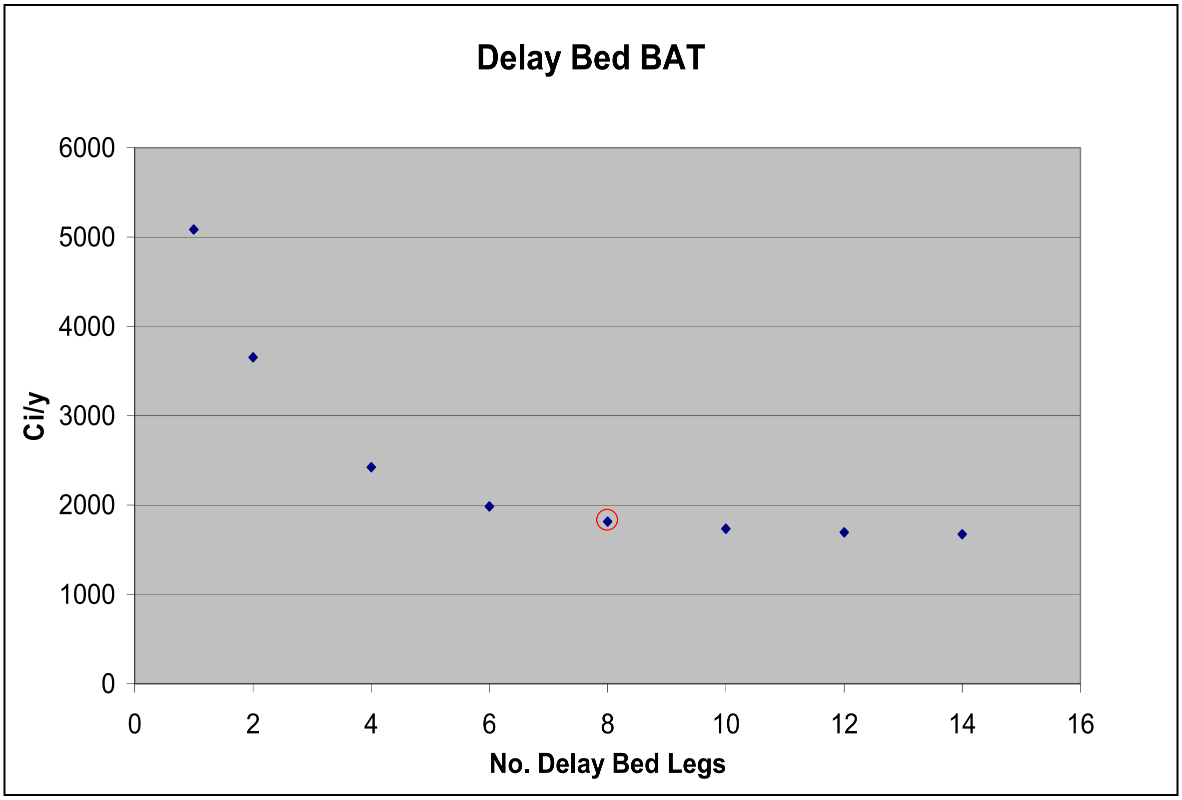
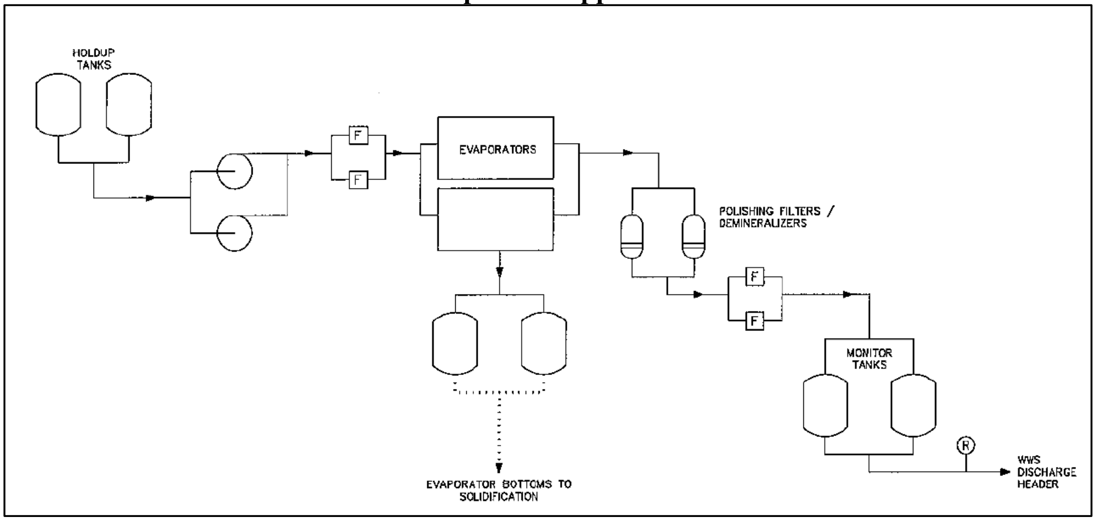


Figure 4-1. BAT Sizing of WGS Delay Beds

Evaporator Approach



Ion Exchange Approach

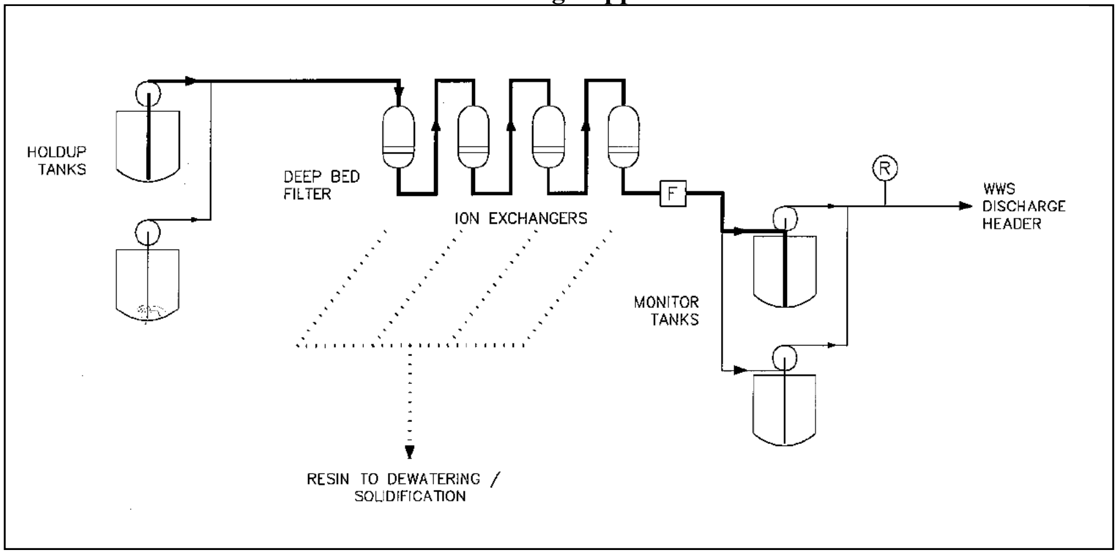


Figure 4-2. Comparison of Evaporator and Ion Exchange Flow Sheets for Liquid Radwaste Treatment

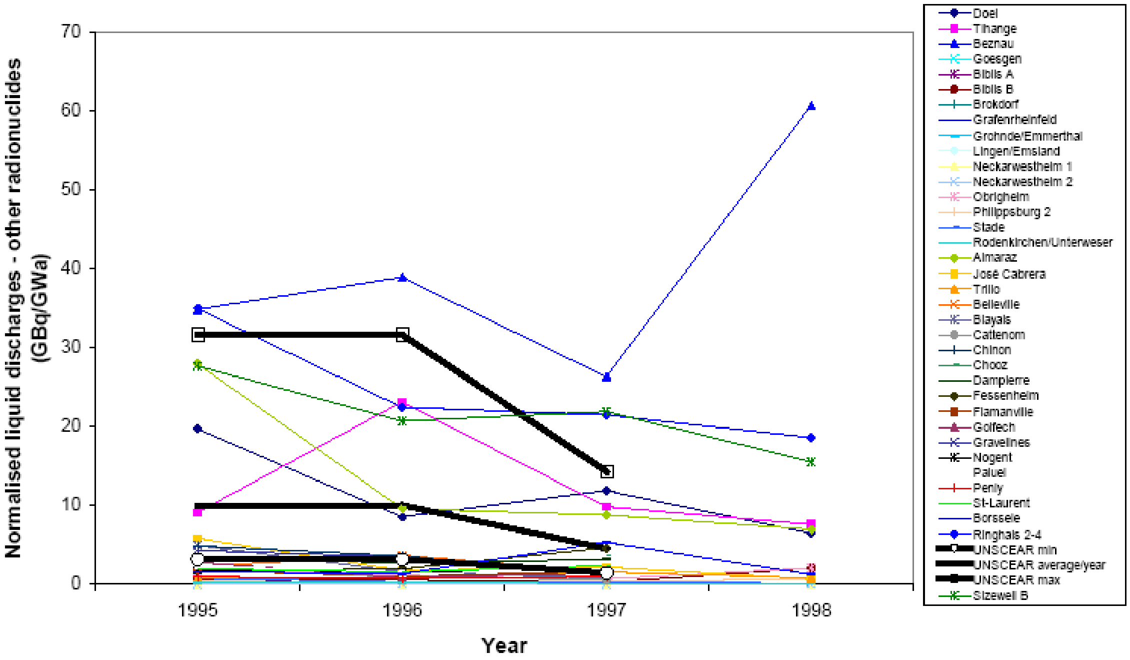


Figure 4-3. Annual Normalised Discharges of Other Radionuclides than Tritium from PWRs 1995-1998 [Reference 10]

5.0 BAT ASSESSMENT – KEY RADIONUCLIDES

5.1 Key Radionuclides

The formation and abatement of key radionuclides has been assessed for the AP1000 NPP. The radionuclides evaluated were selected on the basis of those which:

- are significant in terms of their radiological impact,
- are significant in terms of activity,
- have long half lives and may persist or accumulate in the environment,
- are indicators of plant performance, or
- provide for effective regulatory control.

5.1.1 Key Gaseous Radionuclides

The gaseous isotopes selected for BAT evaluation are listed in Tables 5-1.

The key gaseous isotopes identified in Table 5-1 are: H-3, C-14, N-16, Ar-41, Co-60, Kr-85, Sr-90, I-131, Xe-131m, Xe-133, Cs-137.

5.1.2 Key Radionuclides in Liquid Discharges

The gaseous isotopes selected for BAT evaluation are listed in Tables 5-2.

The key isotopes in liquid discharges identified in Table 5-2 are: H-3, C-14, Fe-55, Co-58, Co-60, Ni-63, Sr-90, Cs-137, Pu-241

5.2 Radionuclide BAT Assessment

The radionuclide BAT assessment involved preparation of a BAT form to identify the radionuclide formation mechanisms in the AP1000 NPP, and the measures taken to minimise production and control releases in gaseous emissions and liquid discharges. The form was developed in consultation with the Environment Agency.

The form identifies the following information:

- Characteristics of the isotope
- Source activity and formation mechanisms
- Techniques to prevent or minimise formation at source
- Pathway to environment
- Downstream abatement techniques – gaseous
- Downstream abatement techniques – liquid
- Emission or discharge data
- Comparison with emissions of discharges from other nuclear power plants
- Dose rate contributions

Various techniques available for the minimisation of the radionuclide formation at source and the abatement of gaseous emissions and liquid discharges are identified in the form and evaluated by a simple scoring system addressing the following criteria:

- Proven technology
- Available technology
- Effective technology

- Ease of use
- Cost
- Impact on public dose
- Impact on operator dose
- Impact on environment
- Generation of a suitable waste form
- Formation of secondary and decommissioning wastes

The scoring system allowed scores of -2, -1, 0, 1 and 2 to be applied to each criterion. A negative score indicates poor performance against the criteria. A positive score indicates good performance against the criteria. A zero score indicates neither a benefit nor a disadvantage. All criteria were considered equally significant (that is, no weighting factor was applied to any criterion to signify a different level of importance). The scores were summed to give a total score for each option.

For the purpose of carrying out the radionuclide BAT assessment, the noble gas isotopes of argon, krypton and xenon were grouped together because they exhibit similar behaviour and have the similar abatement techniques.

The radionuclide BAT assessment of Fe-55, Co-58, Co-60 and Ni-63 was carried out as a grouping with other beta particulate isotopes because they have similar abatement techniques.

The completed radioisotope BAT forms are presented in Appendix A. The scores applied to the minimisation and abatement options were based on professional judgment. The options used in the AP1000 NPP are highlighted in green in the option assessment tables in each form. The radionuclide BAT assessment confirms that the highest scoring techniques for minimisation and abatement have been selected in the AP1000 NPP design.

Table 5-1

SELECTION OF KEY GASEOUS RADIONUCLIDES

Selection criteria	**Isotope**
Significant in terms of their radiological impact	>1% contribution to fisherman family dose (µS/y): C-14, I-131, H-3, Ar-41 >1% contribution to 500y collective dose (manSv) C-14, H-3
Significant in terms of activity	>10% activity (Bq/y): Kr-85, H-3, Xe-131m, Xe-133, Ar-41
Have long half lives and may persist or accumulate in the environment	Half-life >10 years, concentration factors (terrestrial organisms) >1000 and release rates >3.7E+04Bq/y: C-14
Indicators of plant performance	Indicative of particulate emissions: Co-60
Provide for effective regulatory control	Main Vent: Sr-90/Cs-137, I-131, Kr-85/Xe-133 Turbine building vent: Kr-85/Xe-133 Internal vent monitors: Sr-90/Cs-137, Kr-85/ Xe-133, N-16 Grab samples: noble gases, iodine, particulates and tritium
Summary	H-3, C-14, N-16(1), Ar-41, Co-60, Kr-85, Sr-90, I-131, Xe-131m, Xe-133, Cs-137

Note:

1. N-16 detectors are used to detect primary-to-secondary coolant leakage and are located near the steam generator main steam outlet and upstream of the turbine. N-16 has a very short half-life of 7.13 seconds and, as such, is not a suitable isotope for use as a regulatory emission standard to atmosphere.

Table 5-2

SELECTION OF KEY RADIONUCLIDES IN LIQUID DISCHARGES

Selection criteria	**Isotope**
Significant in terms of their radiological impact	>1% contribution to fisherman family dose (µS/y): C-14, Co-60, Co-58, H-3 >1% contribution to 500y collective dose (manSv) C-14, H-3
Significant in terms of activity	>10% activity (Bq/y): H-3
Have long half lives and may persist or accumulate in the environment	Half-life >10 years, concentration factors (aquatic organisms) >1000 and release rates >3.7E+04Bq/y: C-14, Ni-63, Cs-137, Pu-241
Indicators of plant performance	Indicative of corrosion: Fe-55, Ni-63 Indicative of fuel leaks: Cs-137 Other particulates expressed as Co-60
Provide for effective regulatory control	Continuously monitored isotopes: Cs-137 Monitored isotopes grab samples: H-3, Co-60, Sr-90, Cs-137
Summary	H-3, C-14, Fe-55, Co-58, Co-60, Ni-63, Sr-90, Cs-137, Pu-241

6.0 BAT ASSESSMENT ILW AND LLW RADWASTE TREATMENT

6.1 Wastes Processed

The radwaste treatment plant handles ILW and LLW generated in the AP1000 NPP [Reference 6].

6.2 Radwaste Treatment Options

A BAT assessment has been carried out on the radwaste treatment system which addresses the waste activities from the transportation point of the "nuclear island" through to dispatch to the ILW storage prior to disposal or to LLW disposal. The BAT assessment involved Aker Solutions, Different by Design Ltd, and Westinghouse Electric Company, and included representatives from several utilities.

The assumption was made that all reasonable opportunities would be taken for waste minimisation, reuse, and recycling; and, where possible, wastes would be declassified by segregation and cleaning to free release standards. Having made this assumption, the BAT assessment focused on the available technologies for the treatment of LLW and ILW.

Initially, an optioneering process was carried out to identify a set of radwaste treatment options [Reference 11]. A prerequisite was that the options must comply with the following:

- Waste must be treated and handled in accordance with current LLW repository Conditions for Acceptance [Reference 12].

- ILW and LLW containers must meet existing agreed Radioactive Waste Management Directorate (RWMD) specifications [Reference 13].

6.2.1 Initial Option Screening

Initial screening of a range of options was undertaken with an aim of filtering out unworkable or unsuitable options at an early stage. The two criteria that were used for initial screening are listed below:

- Process/waste compatibility (a straightforward "Yes or No"). This assesses the suitability of the option for the treatment of the waste stream and the compatibility of the waste stream with the process.

- Technology availability in the United Kingdom (a scale from 1 to 5). This criterion is essential as an option which is not fully tested in the United Kingdom and is unlikely to yield a licensable design solution within a time scale that is commensurate with the GDA submission. In this scoring scheme, 1 represents a completely novel technology with no full scale application, and 5 represents a fully tried and tested, UK-licensed, widely applied technology. A score of 3 would be a widely available, fully mature but non-UK example.

The potential options were evaluated against their process/waste compatibility for each type of waste and also against technology availability for ILW or LLW. The options were given a colour coding based these attributes (see Table 6-1). Red options were eliminated from the process for further optioneering if they did not meet the requirements from this initial screening. Amber options which show some potential, but not necessarily proven for radioactive waste would only be considered further if fully acceptable (green) options were not available. The outcome of the option screening is shown in Table 6-2 [Reference 1].

The options that survived the initial screening were grouped into potential complete solutions. This was carried out for LLW and ILW.

6.2.2 Evaluation of Screened LLW Treatment Options

The initial option screening exercise for LLW identified the potential complete solutions processes shown in Figure 6-1 [Reference 11].

The complete solutions comprise:

6.2.2.1 Sorting

This allows segregation of waste according to its suitability for the downstream process.

6.2.2.2 Size Reduction

As the LLW is a mixture of wastes, it is difficult to specify the best option at this stage of assessment. All of the size reduction options are low-cost technologies and are considered as potential approaches. Hence, they are not discarded at this stage.

6.2.2.3 Volume Reduction

The option of incineration is omitted from further consideration as it is expected that the adverse public perception of this technology will lead to delays in obtaining licensing. Although controlled oxidation addresses many of the incineration issues, it has not yet been licensed in the United Kingdom. In principle, controlled oxidation presents benefits in reducing the volume of waste which, in turn, leads to higher cost savings. It is recommended that design proposals are flexible to accommodate technologies with better volume reduction such as controlled oxidation once these are fully developed and proven.

This leaves the last option, which is compaction, as the most suitable option.

6.2.2.4 Immobilisation

Immobilisation increases transport weights and volumes requiring disposal and costs more in terms of fuel consumption. As immobilisation is not a required approach of the Condition for Acceptance [Reference 12] for the LLW repository, the selected option is no immobilisation.

6.2.3 Evaluation of Screened ILW Organic Resin Treatment Options

The potential complete solutions that passed screening for ILW organic resin are shown in Figure 6-2 [Reference 11].

To evaluate these options further, a scoring workshop was held on 4th June 2008 with 21 attendees from Aker Solutions, Different By Design (DBD), Westinghouse Electric (WEC), Rolls Royce, Vattenfall, RWE, Ulecia Endessa, and Iberdrola. Table 6-3 shows the set of criteria agreed for the scoring process which included the technical, safety, environmental, and economic aspects [Reference 11]. Each criterion was also given a weighting factor which characterised the relative importance of the issue to the workshop attendees.

The scoring was applied to the available options for the treatment of ILW organic resins [Reference 11]. The results are shown in Table 6-4.

The analysis of the complete solutions for ILW organic resins are summarised below. More details can be found in Reference 11.

6.2.3.1 Dewatering Stage

Table 6-4 shows that no dewatering had the highest total weighted score of 167, but also had the lowest primary waste score. The second highest score was 157 for settling/decanting. Once the consideration was given to the need for dewatering to lower the volume of wastes before undergoing encapsulation, settling/decanting proved to be the most sensible option and was selected for the dewatering stage.

6.2.3.2 Volume Reduction Stage

Table 6-4 shows that no compaction has the highest total weighted score of 170 for volume reduction. Compaction leads to higher cost and introduces additional safety hazard and operability issues. Hence, the option of no compaction is selected.

6.2.3.3 Passivation Stage

Both the solutions of controlled oxidation and wet oxidation are similar in terms of overall benefit, but controlled oxidation is expected to cost more. Although they both can offer benefits in waste reduction, their proven availability is not expected to fall within the GDA submission stage. Hence, no passivation, which received the highest total weighted score of 167, is the selected option.

6.2.3.4 Immobilisation

The option of vitrification is eliminated as it emerges as the most costly and least beneficial option with the lowest total weighted score of 85. Vitrification is also not a well developed and matured technology and is not expected to meet the timeline for the GDA submission stage. The other two options, cement and polymer encapsulation, scored 149 and 128 respectively. Cement encapsulation has the highest score and has the following advantages [Reference 14]:

- This technology is widely used internationally and is well known as a practical and economic approach.
- Radioactive wastes are transported safely.
- This technology meets requirements for RWMD-compliant package [Reference 13].
- This technique has very high reliability of physical containment. The estimated life span is believed to be more than 1000 years. It also allows 97 percent of radionuclides to decay in-situ.
- The porous structure of the cement in this technology enables gas generated from anaerobic conditions and microbial degradation to be emitted from waste packages. This helps in de-pressurisation of the system.
- High pH conditions provided by cement which generates (OH^-) ions will create a barrier against solubility. Soluble radionuclides present in wastes will react with high pH water to form oxides or hydroxides which are insoluble. Hence, migration or transport of radionuclides is reduced.

6.2.4 Cost/Benefit Analysis of ILW Organic Resin Radwaste Treatment

Further analysis on capital cost has been carried out to determine the feasibility of the ‘complete’ solution [Reference 11]. Figure 6-3 shows that simple encapsulation options require least capital cost compared to wet oxidation and controlled oxidation.

Over the lifetime of disposal, the costs of disposal outweigh the capital costs of waste treatment equipment. Figure 6-4 shows that vitrification, wet oxidation and controlled oxidation become more cost effective when predicted lifetime disposal costs are taken into consideration. This is because these technologies result in volume reduction rather than the volume addition associated with encapsulation. However, the necessary development of these technologies is unlikely to happen before the GDA process is complete, but could occur in the future. Therefore, the final selections for ILW resins (organic) radwaste system are the settling/decanting followed by cement encapsulation. It is proposed to use mobile encapsulation facilities on site. This brings the benefit of enabling future technology updates to be integrated into the immobilisation system if a plant operator decides to investigate that. Mobile encapsulation facilities also enable the system to be moved to other locations, increasing its potential for utilisation.

6.2.5 Evaluation of Screened ILW Filter Treatment Options

The potential complete solutions for ILW filter treatment are shown in Figure 6-5 [Reference 11].

The complete solutions for ILW filter treatment comprise:

6.2.5.1 Size Reduction and Volume Reduction

It was preferred that the treatment options for ILW filter is to be of similar with the ILW organic resin treatment options. This is due to the low amount of wastes in this category and also justification on the investment. Hence, this led to the reasoning that neither size nor volume reduction are needed by assuming that the filters can be accommodated within the disposal package without size reduction.

6.2.5.2 Immobilisation

There are no issues with the choice of immobilisation by cement encapsulation, and it has the advantage of being the same process proposed for ILW organic resins.

6.2.6 BAT Radwaste Conclusion

Figure 6-6 summarises the ILW and LLW radwaste treatment options that are selected following the BAT exercise.

For LLW radwaste, the treatment process is based on sorting, sizing (for example, cutting, shredding, and crushing) and compaction.

For ILW radwaste comprising organic resins, the case for dewatering by decantation/settling is strongly argued because of major savings in terms of waste disposal volumes, environmental impact, and cost. Cement encapsulation provides a currently simple, well understood technology that complies with current transportation and waste repository requirements. There are grounds to state that waste disposal volumes and cost may be reduced through the technology development of vitrification or oxidation. However, the development of these technologies is unlikely to happen before the submission of the GDA.

Hence, the final selections for ILW resins (organic) radwaste system are the settling/ decanting followed by cement encapsulation. ILW filters will also be treated by cement encapsulation

6.3 Treatment of Wastes Generated from Radwaste Process

6.3.1 Air

The mobile cement encapsulation plant will be located in the Auxiliary Building. Extraction systems will be connected to the radiologically controlled area ventilation system which vents via the monitored plant vent.

Extract air from the Radwaste Building will be by means of low level extract grilles and conveyed through high integrity ductwork to HEPA filters and discharged to the monitored plant vent by two 100% extract fans.

6.3.2 Water

Water that is decanted from the ion exchange resins will be returned to the ion exchange resin tanks.

6.3.3 Waste

Solid wastes that are generated in the solid waste handling systems will collected and segregated into ILW or LLW streams for processing with other AP1000 NPP solid wastes.

6.4 Comparison of Existing Plant and Best Practice for Solid Wastes

6.4.1 Comparison with Sizewell B

BEGL carried out a review of the control and impact of the discharge and disposal of radioactive waste at Sizewell B in 2005 [Reference 8]. The review was prepared as a submission of information to the Environment Agency to enable its review of Radioactive Substances Act 1993 authorisations. In 2006, the Environment Agency published their decision document and authorisations regarding future regulation of disposals of radioactive waste at UK nuclear power stations [Reference 9]. This review commented on the BPEO and BPM proposed by British Energy for control of radioactive wastes from Sizewell B.

Table 4-15 presents the BPEO issues identified for solid wastes at Sizewell B and a compares them with the practices proposed for the AP1000 NPP. The table also provides a summary of the Environment Agency comments on the Sizewell B BPEO issues.

In general, the proposed AP1000 NPP practice is consistent with practices that were identified as BPEO at Sizewell B. The exception is where on-site incineration was proposed as BPEO. This proposal was not accepted by the Environment Agency. The AP1000 NPP generic design does not have an on-site incinerator.

6.4.2 Comparison with European Practices

The practices at various nuclear facilities within Europe were identified with cooperation of various utilities that participated in the BAT workshop including E.ON, RWE, Endesa, Iberdrola, Suez, and Vattenfall.

Table 6-6 identifies how LLW and ILW solid waste is handled at several European nuclear power plants. More details of the European practices can be found in utility presentations attached in Appendix B.

The examples presented show that the Spanish and Swedish practices for ILW follow a similar cementitious encapsulation approach to that proposed for ILW in subsection 6.2.6. The use of polymeric resin encapsulation is more common in France. The German approach of in-package drying of resin followed by storage does not produce a product that complies with current RWMD waste package specifications. However, the approach does have benefits in reducing total waste volumes and allowing recovery of the dehydrated resin if required. The resin compaction technique employed at Tihange, Belgium also produces smaller waste volumes than cement encapsulation, but the compacted product does not conform to UK conditions for acceptance (CFA) without further conditioning.

The comparison shows a number of different practices for the disposal of ILW in European countries. The cementitious encapsulation option proposed for the ILW generated by the AP1000 NPP is practiced elsewhere in Europe and is consistent with current UK CFA.

Table 6-1

COLOUR CODING REPRESENTATIONS FOR INITIAL SCREENING RESULTS

Colour Coding	**Colour Representations**	
	Waste/Process Compatibility	**Technology Availability(1)**
Red	No	1 or 2
Amber	–	3
Green	Yes	4 or 5

Note

1. Scoring scheme ranges from 1 - 5:
   1 represents a completely novel technology with no full scale application
   3 would be a widely available, fully mature but non UK example
   5 represents a fully tried and tested, UK licensed, widely applied technology

Table 6-2

INITIAL RADWASTE TREATMENT OPTION SCREENING RESULTS [REFERENCE 11]

Processing Option	**Process/Waste Compatibility**						**Technology Availability**		**Comments**
	ILW Resins (organic)	**ILW Resins (inorganic)**	**ILW Charcoal**	**ILW Filters**	**ILW Metal Scrap**	**Mixed LLW**	**ILW**	**LLW**	
Prevent/Reduce	Y	Y	Y	Y	Y	Y	5	5	Essential component in waste management strategy. To be performed at source of waste. Partial solution – Waste consigned to radwaste requires further treatment.
Segregate	N/A	N/A	N/A	N/A	Y	Y	5	5	Assumptions are: 1) Sorting of mixed LLW waste allows for selection of the appropriate treatment(s) for constituent waste streams; 2) Charcoal and resin streams will be treated via the same processes. Therefore, segregation is not required other than dewatering – covered later.
Store as Raw Waste									
Solids	N/A	N/A	N/A	Y	Y	Y	5	5	Unacceptable for disposal. However, may be contingency option if CFA cannot be determined
Solid/liquid mixture	Y	Y	Y	N/A	N/A	N/A	5	5	As for solids above
Volume/Size Reduction									
Size Reduction	N	N	N	Y	Y	Y	5	5	Partial solution only – require further treatment
Compaction/ supercompaction	Y	Y	Y	Y	Y/N	Y	5	5	Final treatment for LLW. ILW would require overpacking. Is a potential viable process for hollow items; for example, tubes, canisters, but not for valves and solid items

Table 6-2 (cont.)

INITIAL RADWASTE TREATMENT OPTION SCREENING RESULTS [REFERENCE 11]

Processing Option	**Process/Waste Compatibility**						**Technology Availability**		**Comments**
	ILW Resins (organic)	**ILW Resins (inorganic)**	**ILW Charcoal**	**ILW Filters**	**ILW Metal Scrap**	**Mixed LLW**	**ILW**	**LLW**	
Non-destructive treatment									
Drying	Y	Y	Y	N	N	N	5	N/A	Partial solution only - require further treatment
Evaporation	N	N	N	N	N	N	5	5	Applicable to liquid wastes only
Dewatering (settling/ decanting)	Y	Y	Y	N	N	N	5	N/A	Partial solution only - require further treatment
Filtration	Y	Y	Y	N	N	N	5	N/A	Partial solution only - require further treatment
Decontamination	N	N	N	Y	Y	Y	5	5	Partial solution - creates secondary wastes, requires further treatment
Absorption	Y	Y	Y	Y	Y	N	5	N/A	Partial solution - requires further treatment. For metal wastes, is limited to swabbing to remove surface water dependent on downstream process selection.
Direct immobilisation	Y	Y	Y	Y	Y	Y	5	5	May require pre-treatment to passivate organics
Destructive treatment									
Conventional incineration	Y	Y	Y	Y	N	Y	2	5	Partial solution passivates waste - requires further treatment to immobilise. No known applications for ILW resins.

Table 6-2 (cont.)

INITIAL RADWASTE TREATMENT OPTION SCREENING RESULTS [REFERENCE 11]

Processing Option	Process/Waste Compatibility						Technology Availability		Comments
	ILW Resins (organic)	**ILW Resins (inorganic)**	**ILW Charcoal**	**ILW Filters**	**ILW Metal Scrap**	**Mixed LLW**	**ILW**	**LLW**	
Controlled oxidation	Y	N	Y	Y	N	Y	3	3	Partial solution - requires further treatment to immobilise. Could be used on inorganic IX resin; however, provides no benefit. No UK applications; several in U.S. and Europe
Vitrification	Y	Y	Y	N	N	Y	4	2	Single UK application on liquid HLW; several application worldwide, including other wastes; limited use for LLW
Synroc	Y	Y	Y	N	N	Y	2	2	Developed for liquid HLW; mainly used for High Pu military wastes. No UK application
Plasma arc	Y	Y	Y	Y	Y	Y	2	2	Either with frit to form of glass or without - without requires further treatment of ash (that is, encapsulation). No full scale nuclear application in UK or elsewhere
GeoMelt	N	Y	N	N	N	N	2	N/A	Only known applications are in the ground and non-UK.
Molten-salt oxidation	Y	Y	Y	N	N	Y	2	2	Partial solution only - requires further treatment. Emergent technology - lab scale only
Wet air oxidation (WETOX)	Y	N	Y	N	N	N	4	N/A	One UK licensed mobile plant. Partial solution only - require further treatment

Note

ILW resins (organic), ILW resins (inorganic) and ILW charcoal will be treated as the same waste stream

Table 6-3

MAPPING OF SCORING REQUIREMENTS AGAINST CRITERIA [REFERENCE 11]

Criterion		Weight	Score					Description
			1	2	3	4	5	
Technical	Technology availability	4	Essentially a completely novel and unproven concept. No evidence of nuclear industrial/commercial application. Considerable fundamental development work anticipated to bring to UK licensable position	Novel concept which has undergone a significant amount of development to underpin its feasibility. Little/no evidence of full-scale deployment either in United Kingdom or elsewhere although pilot scale plants may exist. Major effort needed to develop to a deployable condition and to establish UK licence position.	Evidence of technology deployment in nuclear industrial/commercial applications overseas. Potentially viable for UK-use. However, significant effort is anticipated to secure UK licensing.	Evidence of UK deployment, although limited examples exist currently. Licensable technology, although a moderate amount of work is anticipated in ensuring its application to this project.	Many examples of technology application in UK industry. Well documented process. Little/no problems anticipated with UK licensing	This assesses the maturity of the technology being considered and reflects the uncertainty of whether the option will be successful and, therefore, the amount of development required to underpin an option and enable its successful implementation. A low score will be earned where the technology remains to be proven (that is, will it work?) or developed (how well will it work?). A tool such as the Technology Evolution Index (TEI) can be used as a measure. This attribute is focused on technical confidence. The time to undertake development work is addressed under the implementation time attribute.
	Operability/ maintainability	4	Highest complexity; highest potential for outages. Lowest overall availability	Highly complex; high potential for outages. Low overall availability	Moderately complex; moderate potential for outages. Moderate overall availability	Low complexity; low potential for outages. High overall availability	Lowest complexity; lowest potential for outages. Highest overall availability	An assessment of the inherent availability, reliability and maintainability. At the stage of development of the option, this will be based on a view of the scope and complexity of the envisaged plant. A complex heavily engineered plant or one with a large number of process steps will increase the likelihood of maintenance periods, reducing the overall availability. Concerned with plant availability as distinct from technology availability
Safety	Dose uptake	4	Highest complexity; highest potential for outages and hands-on activities. Highest potential for non-routine dose uptake	Highly complex; high potential for outages and hands on activities. Highest potential for non-routine dose uptake	Moderately complex; moderate potential for outages and hands on activities. Moderate potential for non-routine dose uptake	Low complexity; low potential for outages and hands on activities. Low potential for non-routine dose uptake	Lowest complexity; lowest potential for outages and hands on activities. Lowest potential for non-routine dose uptake	As a new facility built to modern plant standards, routine dose uptake is not likely to be a major discriminator. The potential for radiation exposure will be most likely to occur during periods of manual intervention for maintenance during breakdowns and then will be designed to stay within target levels. However, the potential for dose uptake will increase with the frequency and occupancy of maintenance episodes. At a conceptual stage, it will be judged as a function of the scope and complexity of the process.

Table 6-3 (cont.)

MAPPING OF SCORING REQUIREMENTS AGAINST CRITERIA [REFERENCE 11]

Criterion		Weight	Score					Description
			1	2	3	4	5	
Safety	Hazard potential (radiological)	4	High no. of high consequence potential accident scenarios. Very difficult to design out. Very heavy reliance on active engineered protection	High no. of/or high consequence potential accident scenarios. Difficult to design out. Heavy reliance on active engineered protection	Medium no./ consequence of potential accident scenarios. Some reliance on engineered protection	Low no. of potential accident scenarios; mostly easy to design out. Low consequence. Minimal engineered protection	Inherently safe. Very low no. of potential accident scenarios. Very low consequence	To address the radiological hazard potential (frequency and consequence) from reasonably foreseeable accident scenarios of each option and the confidence that hazards can be managed to achieve national risk criteria. It reflects the option's potential for management of radiological hazards against the Hazard Management Hierarchy of Eliminate, Prevent, Mitigate, Protect- passive means, Protect - active means. That is, an option that is inherently safe will score more highly than one that places heavy reliance on engineered protection.
	Hazard potential (non-radiological)	3	High no. of high consequence potential accident scenarios. Heavy reliance on managerial control and protective measures	High no. of /or high consequence potential accident scenarios. Significant reliance on managerial control and protective measures	Medium no./ consequence of potential accident scenarios. Moderate reliance on managerial control and protective measures	Low no. of potential accident scenarios. Low consequence. Some reliance on managerial control and protective measures	Inherently safe. Very low no. of potential accident scenarios	A measure of the option's performance in management of conventional safety hazards (temp. pressure, height, confined space, moving machinery, and the like). An option that is inherently safe will score more highly than one that places heavy reliance on protection measures or managerial/supervisory control. Considers construction, operation and decommissioning
Environmental	Primary waste management	5	Considerable increase in primary waste volumes	Significant increase in primary waste volumes	No or insignificant reduction or increase in primary waste volumes	Significant reduction in primary waste volumes	Considerable reduction in primary waste volumes	A measure of the option's potential performance in the management of primary wastes. Considers Waste Management Hierarchy Principles of Prevent, Reduce, Reuse, Recycle, Recover, Dispose while recognising that prevention occurs at source and, therefore, focuses on reduction or conversely additional waste generation through the treatment process. For the purposes of the scoring exercise, primary waste is classed as the combined volume of resin and water crossing the system boundary into radwaste treatment. Water: solids taken as ~ 1:1 v/v
	Secondary waste management	4	Considerable and/or problematic secondary wastes (solid, liquid, gaseous) generated	Significant amounts of secondary wastes generated requiring a secondary/subsidiary process route.	Significant amounts of secondary wastes generated requiring secondary/subsidiary process route.	Moderate amounts of secondary wastes generated requiring a secondary/subsidiary process route.	Minimal to no secondary wastes generated as a result of the specific process proposed.	A measure of the option's potential performance in the management of secondary wastes. Secondary wastes to be taken as including S.LG waste streams including new liabilities and consumables; for example, filters or other media. Does not consider generic effluents; for example, washdown that is common to all options
	Planning issues	2	Very high probability of inquiry. Long delays to consent envisaged	High probability of inquiry.	Moderate probability of inquiry.	Low probability of inquiry.	Very low probability of inquiry. No extra ordinary delays to consent envisaged	This reflects the probability of delays through planning issues; for example, with respect to public inquiry, and is particularly relevant to options such as incinerators.

Table 6-3 (cont.)

MAPPING OF SCORING REQUIREMENTS AGAINST CRITERIA [REFERENCE 11]

Criterion		Weight	Score					Description
			1	2	3	4	5	
Environmental	Product quality	5	Very low confidence in meeting current UK specs	Significant uncertainty regarding whether technology proposed would ever meet UK specs. Meets only isolated conditions or achieves partial compliance on all conditions	Could be made to meet UK specs only by the addition of a complementary process. Meets ~50% of conditions as a standalone process	Nearly meets all requirements e.g. Meets most CFA fully with partial compliance on isolated conditions. May be granted an Letter of Compliance if it can be demonstrated that all reasonable measures have been taken	Very high confidence in meeting current UK requirements. Fully meets all CFA	An indication of the option's potential to produce a product that gains a Letter of Compliance from RWMD by meeting their CFA for the ILW Repository: immobilised, free of water, homogeneous, radiologically stable, chemically passive (that is, zero gas generation), characterised, voids minimised (Reference 2, Nirex GWPS Vol 2.) Alternatively to meet CFA for LLW repository in the case of mixed waste /trash
	Resource usage	1	Very high resource demand	High resource demand	Moderate resource demand	Low resource demand	Very low resource demand	To compare the relative potential consumption of resources (non-human), including raw materials, water, energy. Does not consider demand for human resources which is covered under operational costs
Economic	Implementation Time	2	Time to develop design to appropriate standard for GDA submission is well beyond deadline.	Time to develop design to appropriate standard for GDA submission is behind deadline.	Time to develop design to appropriate standard for GDA submission is on deadline.	Time to develop design to appropriate standard for GDA submission is within deadline.	Time to develop design to appropriate standard for GDA submission is well within deadline.	Time to implement the Radwaste Building is unlikely to be a factor relative to the time to implement the reactor plant. Therefore, the time to submit designs relative to the GDA deadline is used as the benchmark instead.
	Process technology costs	3	Highest overall relative cost. Substantial investment anticipated in fundamental research and development. Greatest scope, most complex process. Greatest operator demand	High relative cost for the technology option. Expected to require significant development cost. High scope, complex process. High operator demand	Medium relative cost. Moderate scope, moderately complex process. Moderate operator demand	Low relative cost. Low scope, fairly simplistic process. Low operator demand	Lowest relative cost. Least scope, simplest process. Least operator demand	A relative assessment of treatment costs includes development, design, capital & operating costs. At an early stage, the score will reflect the anticipated scale, scope, and complexity of the process plant rather than a full engineering estimate against bill of quantities, rates and norms.

Table 6-4

SUMMARY RESULTS OF OPTION SCORING [REFERENCE 11]

		Technical Criteria			Safety Criteria			Environmental Criteria			Economic Criteria			
		Technology Availability	Operability/ Maintainability	Dose Uptake	Hazard Potential (Radiological)	Hazard Potential (Non-Radiological)	Primary Waste Management	Secondary Waste Management	Planning Issues	Product Quality	Resource Usage	Imple-mentation Time	Process Technology Costs	
Option Set	Weight / Option	4	4	4	4	3	5	4	2	5	1	2	3	Total Weighted Score
De-Water	None	5	5	5	5	5	3	5	5	N/A	5	5	4	167
	Drying	5	3	3	3	3	4	4	5	N/A	3	5	4	136
	Absorption	5	4	4	4	4	3	5	5	N/A	4	5	4	151
	Settling / decanting	5	4	4	4	4	4	5	5	N/A	5	5	4	157
	Filtration	5	4	4	4	4	4	5	5	N/A	4	5	4	156
Volume Reduction	None	5	5	5	5	5	3	5	5	N/A	5	5	5	170
	Compaction	5	3	3	3	3	3	5	5	N/A	4	3	3	129
Passivation	None	5	5	5	5	5	3	5	5	N/A	5	5	4	167
	Controlled oxidation	3	2	2	2	2	5	4	3	2	3	3	1	111
	WETOX	4	2	2	2	3	5	3	4	2	3	2	2	117
Immobilisation	Polymer encapsulation	4	3	3	4	2	2	4	5	2	3	4	3	128
	Vitrification	2	1	1	2	2	5	3	3	1	2	1	1	85
	Cement encapsulation	5	3	3	4	4	2	4	5	4	4	4	3	149

Table 6-5

SUMMARY OF TREATMENT OF LLW AND ILW SOLID WASTES AT VARIOUS NUCLEAR POWER PLANTS IN EUROPE

Country	Belgium	France	Germany	Spain	Sweden
Plant / Waste	Tihange [Reference 15]	EDF	RWE - Various	Iberdrola -Various	Ringhals
LLW		Compaction	Compactable: • Supercompaction and packaging into repository accepted bin (drum/box) Non-compactable: • Packaged in accepted bin (drum/box). • Burnable wastes incinerated. Slag, dust and ash stored in a tank and packaged in 200-I-drums. Metallic: • Ferrous and non-ferrous metals melted, re-used where possible, packaged/ compacted into 200-I-drums. • Copper cables/wiring shredded, recycled where possible, and packaged in 200-I-drums.	Compactable: • In drum (220 litre) compaction Non-compactable: • Placement in 220 litre drum + void filling with grout	• Sorted in the unit in special environmental stations • Collected by a special vehicle and transported to the waste treatment building • Categorised measured and registered. • Compacted to bales • Stored in containers awaiting shipment to final repository or deposited in the shallow burial located on-site depending on doserate level and content of nuclides.

Table 6-5 (cont.)

SUMMARY OF TREATMENT OF LLW AND ILW SOLID WASTES AT VARIOUS NUCLEAR POWER PLANTS IN EUROPE

Country	Belgium	France	Germany	Spain	Sweden
Plant / Waste	Tihange [Reference 15]	EDF	RWE - Various	Iberdrola -Various	Ringhals
ILW resins	• Dewatering • Transfer to 200 litre drums • Transfer to drying/heating (thermal oil) vessel • Transfer to metal press drums and lidded • High energy compaction • Pellets packed in 200 litre drums • Interim store or final repository	• Dewatering • Cement or Polymer immobilization • Concrete cask	• In-package drying • Vapour condensate treatment • Storage in MOSAIK cask	• Dewatering • Cement immobilization in 220 litre drum	• Solidified in cement in sheet-metal moulds • Intermediate on-site storage • Transfer to final repository
ILW filters		• Concrete enclosures	• Cementation in stainless steel drums	• Immobilisation in type-filter drum with 5cm concrete wall	
Waste form meets Nirex/ RWMD generic waste package specification	No	Yes	No	Yes	Yes

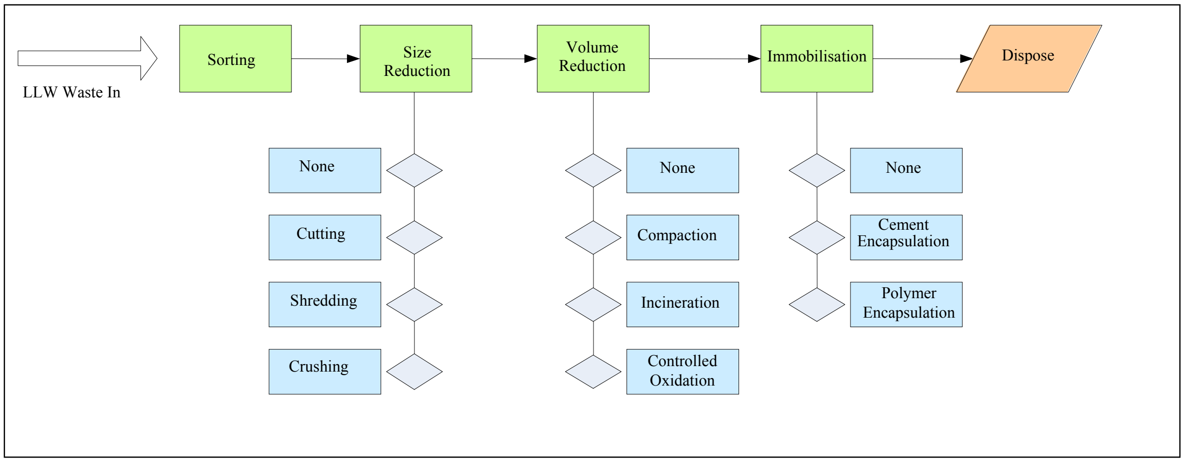


Figure 6-1. Low Level Waste Options

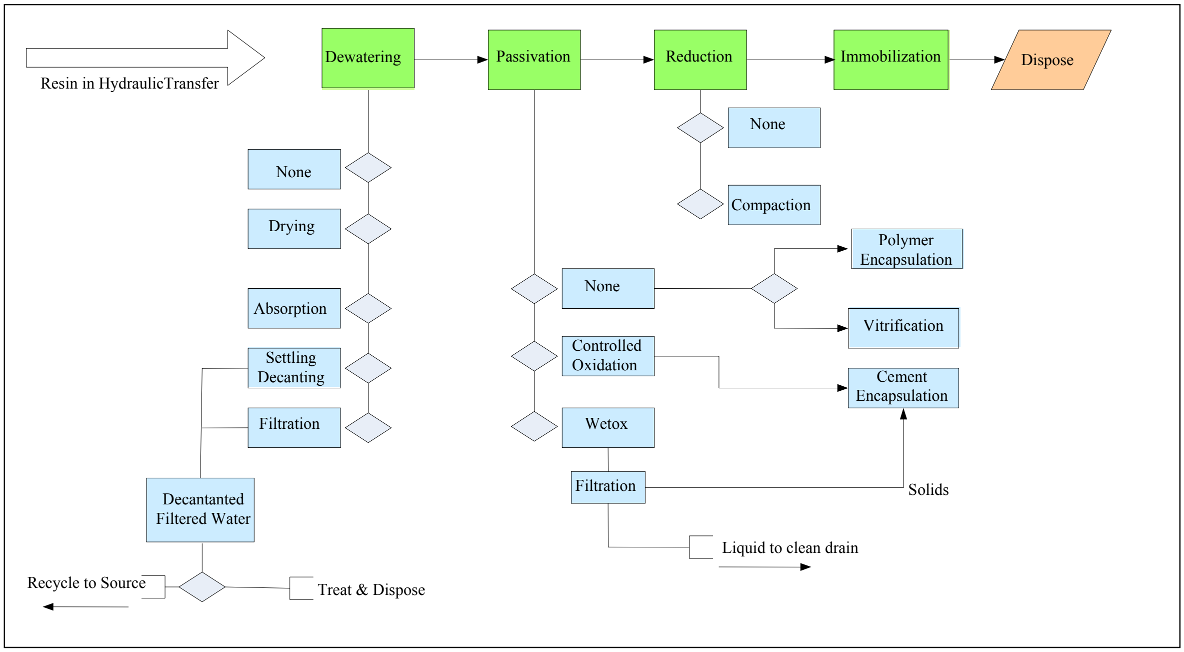


Figure 6-2. ILW Organic Resin Treatment Options

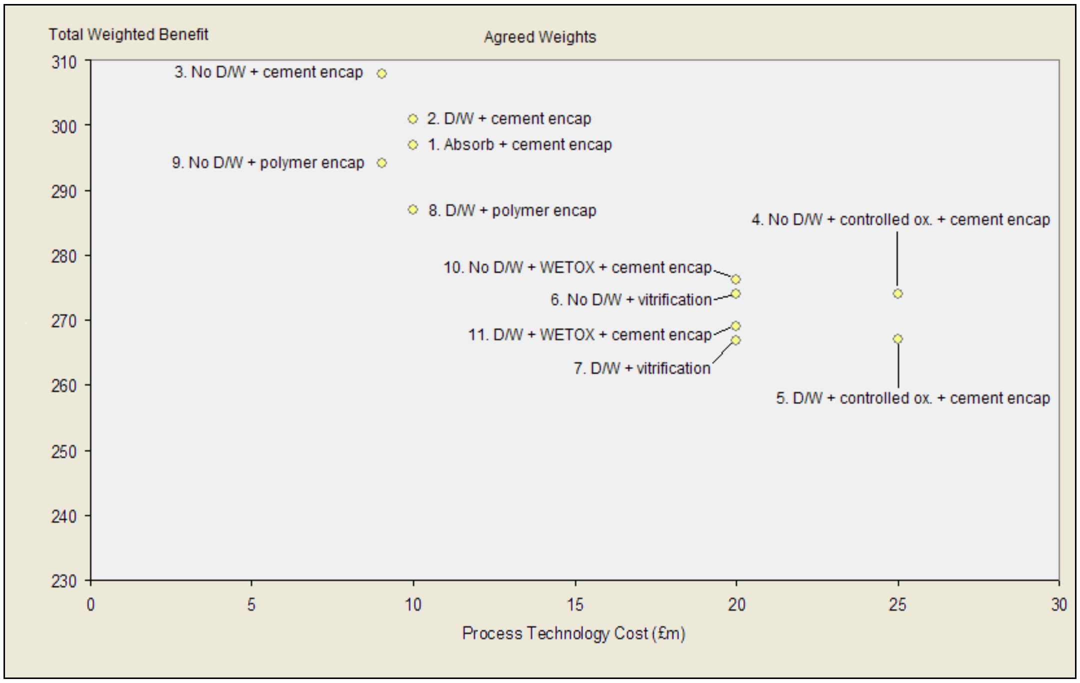


Figure 6-3. Total Weighted Benefit versus Cost of Process Technology [Reference 11]

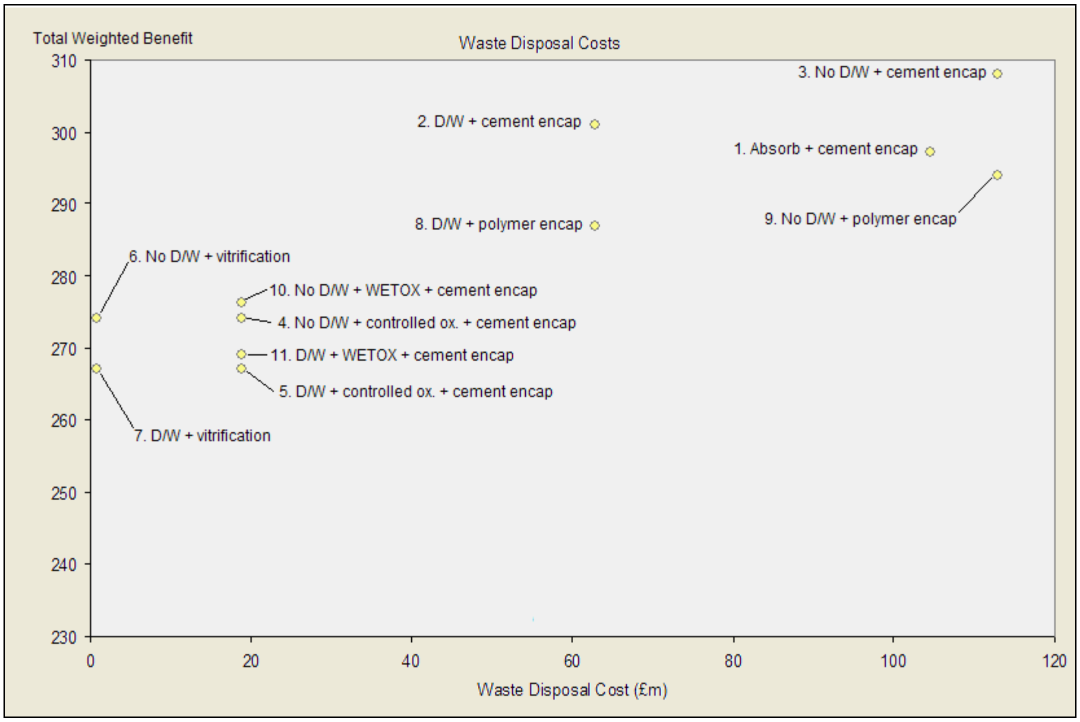


Figure 6-4. Total Weighted Benefit versus Cost of Waste Disposal [Reference 11]

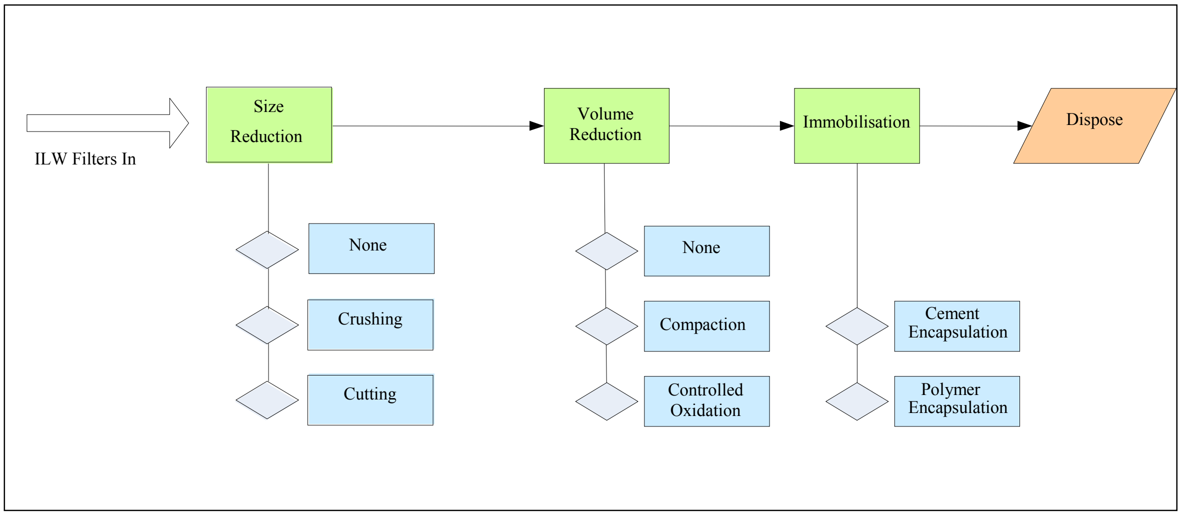


Figure 6-5. ILW Filter Treatment Options

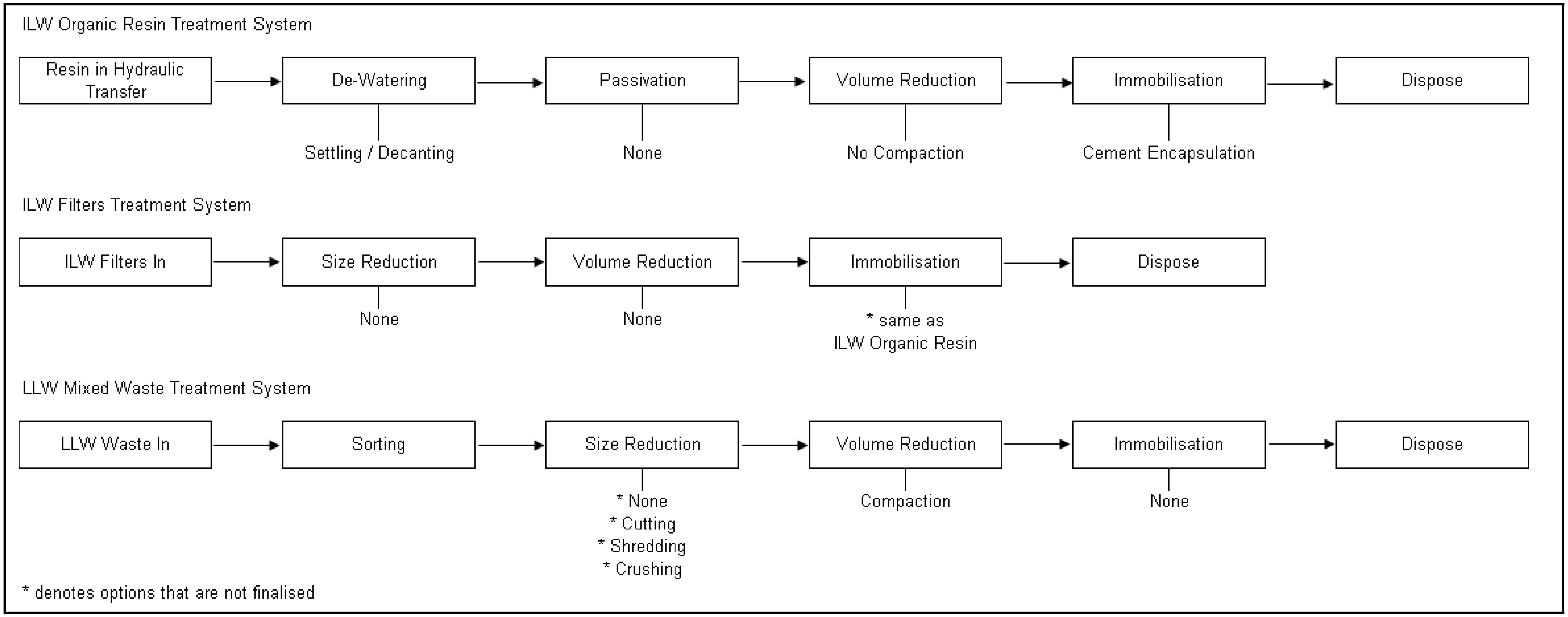


Figure 6-6. Summary of Selected BAT for ILW and LLW Radwaste

7.0 SUMMARY

This report demonstrates how the AP1000 NPP has been designed with the following fundamental design objectives:

- Safety
- Constructability
- Reliability
- Operability
- Maintainability
- Minimisation of radioactive emissions, discharges and waste

These design principles and implementation of ALARA have resulted in a nuclear island design that is equivalent to BAT. The generation of radioactive waste is minimised through design. There are many examples, including the materials of construction (limited use of cobalt-based alloys), canned reactor coolant pumps, mechanical shim control (gray rods), and zinc addition to the RCS. The selection of BAT has been substantiated by the preparation of radionuclide BAT forms which identify the production mechanisms, minimisation techniques, and abatement options for the key radionuclides. The highest scoring techniques have been adopted in the AP1000 NPP design.

The treatment system for gaseous radioactive emissions follow well tried and tested techniques of moisture removal and activated carbon guard beds and activated carbon delay beds.

The treatment system for liquid radioactive waste involves the use of ion exchange beds with a waste prefilter upstream and a waste after-filter downstream of the ion exchangers. A high degree of flexibility is built into the system allowing storage, recirculation to ion exchange, and treatment by mobile plant if required. Treated effluent is stored in monitor tanks and compliance with discharge limits is confirmed before release to the environment.

The treatment of ILW and LLW has been subject to a detailed BAT assessment. The selected techniques are constrained by the requirements to comply with the current LLW repository CFA. The selected techniques also produce a waste form that meets Nirex/RWMD Generic Waste Package Specification [Reference 13] as these are perceived to have the highest chance of meeting future CFA at any ILW repository. The assessment concluded that BAT for ILW is the cement encapsulation, long-term storage, and disposal to the national repository. The BAT treatment and disposal of LLW involves segregation, size reduction and compaction. LLW will be transferred directly into approved waste containers for transfer to the repository. Where possible, waste which can be reclassified as non-radioactive following segregation or decontamination (for example, metals) will be subject to recycling, reuse, or transfer to licensed off-site facilities for the disposal of non-radioactive waste.

Examples of how the four key BAT management factors for the optimisation of releases from nuclear facilities (use of low waste technology, efficient use of resources, reduced emissions, use of less hazardous substances) are implemented in the AP1000 NPP are summarized in Table 7-1.

The AP1000 NPP is a modern design which incorporates the principles of ALARP and BAT to ensure the safety of the public and power plant workers, and minimises the impact of the plant on the environment.

Table 7-1

NUCLEAR BAT MANAGEMENT FACTORS AND AP1000 NPP FEATURES

Use of low waste technology	Efficient use of resources	Reduced emissions	Use of less hazardous substances
• Minimise the generation of radioactive wastes from the nuclear facility *Selection of materials, water chemistry improvements; for example, zinc addition for corrosion control, use of ^{7}LiOH reduces ^{3}H generation, improved fuel performance and higher burn-up. Reduction in the number of components, equipment, materials used to construct. Renovate & reuse where possible; for example, PPE/metal component. Run primary filters and IX media to ILW to minimise waste volumes; use compactable materials for "disposable" components*	• Improve the eco-efficiency of the nuclear facility (for example, emissions/GWa) *Best industry practise and adherence to IAEA guidelines minimising generation of waste. Reduced activated corrosions products. Reduced and simplified equipment inventories. Use of mechanical shims for reactivity control reduces liquid effluents and on-site chemical inventory.*	• Concentrate and contain environmentally persistent or bio accumulative emissions *Use of filtration to capture airborne particulate emissions into solid phase. Select materials that minimise the creation of persistent wastes. Reduction in containment service penetrations. HEPA filter selection*	• Radioactive waste should be created in a passively safe waste form. *Spent fuel stored in racks in the fuel pond at subcritical distances and incorporating neutron absorbing panels (Metamic); spent fuel packed into storage containers for placement in on-site store; utilising passive systems for cooling and containment; storage canisters incorporate neutron absorbing materials. Solid ILW is immobilised in grout (robust mix determined from formulation trials).*
• Radioactive wastes should be created in a manageable waste form. *ILW resins can be pumped (entrained in water). ILW and LLW will be processed and packaged to meet the CFA for the respective waste repositories and utilising RWMD-compliant packages; for example, 3m^3 box/drum*	• Optimise both radioactive and non-radioactive impacts to reduce the environmental footprint of the facility *Use of canned coolant pumps eliminates seal leaks, and creation of waste, bunds, and collection sumps are incorporated to locally retain leaks and spill. Reduced and simplified equipment inventory (less maintenance and decommissioning waste), Reduced building volumes (less decommissioning waste)*	• Reduce transboundary geographic displacement of environmental impacts *ILW is collected, processed, and stored on-site within the site boundary and with suitable shielding in ILW storage vault. Spent fuel is stored on-site within the site boundary and with suitable shielding (fuel pond or on-site store).*	• Condition and immobilise unstable waste forms into a passively safe state *Structurally unstable ILW resins are immobilised in grout (robust mix determined from formulation trials).*
• Minimise treatment and conditioning necessary to safely store wastes *Incinerate suitable LLW (oils, solvents and resins; cementitious encapsulation of ILW resin (without pre-conditioning)*	• Prioritise environment expenditure to maximise the amount of radioactive pollution avoided for each € invested *IX resins capture radionuclides from the soluble to the solid phase in a compact and energy efficient manner, that is, no evaporator used. Use of mechanical shims for reactivity control reducing boric acid use and associated liquid effluent due to reductions in primary circuit liquid volume changes.*	• Minimise potential radioactive releases from credible accident conditions and consequences for the environment *Sealed containment and shield around reactor pressure vessel, catalytic hydrogen recombiners in the containment ventilation system, trisodium phosphate in basket's use of canned coolant pumps eliminates seal leaks and creation of waste, bunds, and collection sumps to locally retain leaks/spills.*	• Wastes should be capable of interim safe storage prior to final disposal in a repository. *ILW store will be constructed on site. All ILW waste packages produced are compatible with the store, associated equipment, and the capability of the shielded vault. Spent fuel will be packaged into suitable storage container for placement into the on-site spent fuel store.*
	• Progressively reduce worker doses from waste treatment and conditioning processes *Reduction in waste volumes (including decommissioning) over previous designs so less time spent handling waste. Suitable shielding and remote handling equipment are incorporated into the designs.*	• Progressively reduce emissions. Ongoing update of management procedures to ensure best industry practice. Reassessment of BAT during plant upgrades to ensure incorporation of latest techniques	• Wastes should be capable of being stored in a monitorable and retrievable waste form. *ILW waste packages are RWMD-compliant. ILW store handling equipment and package inspection bays allow individual packages to be retrieved and monitored. Proposed spent fuel store allows in-situ monitoring; all spent fuel canisters can be retrieved from the on-site store.*

8.0 REFERENCES

1. "Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs," Version 1, Environment Agency, 11th January 2007.
2. "Implementation of PARCOM Recommendation 91/4 on Radioactive Discharges," OSPAR Commission, 2003.
3. EPS-GW-GL-700, Rev 1, "AP1000 European Design Control Document," Westinghouse Electric Company, 2009.
4. APP-GW-GER-005 Revision 1, "Safe and Simple: The Genesis and Process of the AP1000 Design," Westinghouse Electric Company, 2008.
5. EPS-GW-GL-700, Rev 4, "UK AP1000 Environment Report," Westinghouse Electric Company, 2011.
6. UKP-GW-GL-027, Rev 2, "Radioactive Waste Arisings, Management and Disposal," Westinghouse Electric Company, 2011.
7. UKP-GW-GL-033, Rev 2, "Assessment of Likely Impact of Radioactive Discharges on Non-Human Species," Westinghouse Electric Company, 2011
8. "Submission of information to the Environment Agency to enable review of Radioactive Substances Act 1993 Authorisations, Sizewell B Power Station," British Energy Generation Ltd., Report No. SZB/THR/042, July 2005.
9. "Decision document and authorisations for future regulation of disposals of radioactive waste under the Radioactive Substances Act 1993 at British Energy Generation Limited's nuclear sites: Dungeness B Power Station, Hartlepool Power Station, Heysham 1 Power Station, Heysham 2 Power Station, Hinkley Point B Power Station and Sizewell B Power Station," Environment Agency, 21st December 2006.
10. "Implementation of PARCOM Recommendation 91/4 on Radioactive Discharges," OSPAR Commission 2003, ISBN 1-904426-21-2.
11. UKP-GW-GL-039, Rev 0 "Radwaste Treatment Options Study Report," Westinghouse Electric Company, 2009.
12. "Conditions for Acceptance by LLW Repository Ltd. of radioactive Waste for Disposal at the Low Level Waste Repository (CFA)," LLW Repository Ltd, Issue 01/08, Feb 2008.
13. "Generic Repository Studies – Generic Waste Package Specification," Volume 1 – Specification, Nirex Report N/104, United Kingdom Nirex Limited, June 2005.
14. "Why a Cementitious Repository?" Nirex Report N/034, United Kingdom Nirex Limited, June 2001.
15. Andreas Roth, Baudouin Centner, Alain Lemmens: "Radioactive Spent Resins Conditioning by the Hot Super-Compaction Process," Proceedings of the 11th International Conference on Environmental Remediation and Radioactive Waste Management ICEM2007, September 2-6, 2007, Oud Sint-Jan Hospital Conference Center, Bruges, Belgium.

16. "The National Environmental Quality Standard for List II Substances, EPR H1 Environmental Risk Assessment Part 2 Assessment of Point Source Releases and Cost-Benefit Analysis," Environment Agency, 2008.

17. "Rolling summary of the current components of the radioactive waste management policy in the UK," Radioactive Waste Policy Group (RWPG), September 2007.

18. "HI-STORM 100 and HI-STAR 100, Holtec's Systems for Storage and Transport of Spent Nuclear Fuel," Holtec International, Marlton, NJ, U.S.A.

19. An Aid to the Design of Ventilation of Radioactive Areas, NVF/DG001, Issue 1, Nuclear Industry Safety Directors Forum, January 2009.

APPENDIX A
BAT FORMS FOR IMPORTANT RADIONUCLIDES

1. Tritium
2. Carbon-14 (C-14)
3. Nitrogen-16 (N-16)
4. Strontium-90 (Sr-90)
5. Iodine-131 (I-131)
6. Caesium-137 (Cs-137)
7. Plutonium-241 (Pu-241)
8. Noble gases
9. Beta particulates

1. BAT Assessment Form – Tritium

Radionuclide Information

Radionuclide	H-3	**Discharge Route (Solid/Liquid/Gas)**	Gas/Liquid
Radioactive Half-life	12.4 years	**Parent** **Daughter**	N/A He-3
Principal Decay Mode	Beta	**Average Energy**	0.00568 MeV

Speciation	Hydrogen occurs freely in nature as H2, but combines with most elements to form hydrides. In the environment, water is by far the most important hydrogen-containing compound. Hydrogen is a major component of most organic molecules and thus tritium can exchange with hydrogen-1 and become bound to such molecules. [Reference 1, p. 80] Tritium is produced by several reactions in operating reactors. It is readily incorporated with water molecules as HTO and, very rarely, as T_2O. [Reference 4]
Source of Radioactivity (back to the point of generation)	Tritium is one of the most abundant radionuclides present in the coolant. Tritium arises mainly from ternary fission of the uranium fuel followed by transmission through the fuel pin cladding into the RCS. Ternary fission is where the uranium nucleus splits into 3 fragments rather than 2 (this occurs in around 1 in 400 cases where one such fragment may be tritium – [Reference 4]). The production rate for tritium thus depends mainly on reactor power. The transmission mechanisms for tritium through the cladding are generally considered to be a) diffusion of tritium atoms around grain boundaries and through intact cladding, b) effusion of tritium through minute holes or defects in the cladding material, and c) direct penetration of tritons. Hydrogen is the only element small enough to diffuse through the fuel clad even in the absence of fuel defects. The rate of diffusion depends on temperature, which depends on reactor power. Therefore, overall the quantity of tritium released into the coolant is dependent on reactor power. [Reference 2, p. 25] Tritium is also produced by activation of Boron-10, Deuterium (Hydrogen-2), Lithium-6 and Lithium-7: B-10 + n → 2(He-4) + T-3 D-2 + n → T-3 Li-6 + n_{th} → T-3 + He Li-7 + n → He-4 + T-3 + n The variation of these isotopes in the reactor coolant throughout the 18-month fuel cycle is responsible for the greatest variation in the tritium production rate. The reactor coolant is almost entirely H_20 molecules (normal water) with a small proportion of tritiated water. [Reference 2, p. 25]

1. BAT Assessment Form – Tritium (cont.)

Source Activity (Bq)

(before abatement)

Tritium Sources:

	Release to Coolant **(curies/cycle [TBq /cycle] – 18 month cycle)**	
Tritium Source	Design Basis (10% in-core tritium released to coolant)	Best Estimate (2% in-core tritium released to coolant)
Produced in core		
Ternary fission	1770 [62.9]	354 [13.098]
Burnable absorbers	279 [10.323]	56 [2.072]
Produced in coolant		
Soluble boron	734 [27.158]	734 [27.158]
Soluble lithium	168 [6.216]	168 [6.216]
Deuterium	4 [0.148]	4 [0.148]
TOTAL	2955 [109.335]	1316 [48.692]

[Reference 3, Table 11.1-3]

Design Basis

Steam Generator Secondary Side Liquid Activity: 1.0 μCi/g [37,000 Bq/g]

[Reference 3, Table 11.1-5]

Realistic Source Terms

Reactor Coolant Activity: 1 μCi/g [37,000 Bq/g]

Steam Generator Liquid Activity: 1.0×10^{-3} μCi/g [37 Bq/g]

Steam Generator Steam Activity: 1.0×10^{-3} μCi/g [37 Bq/g]

[Reference 3, Table 11.1-8]

Total secondary side water mass in Steam Generators – 3.5×10^{5} lbs [158.757×10^{3} kg]

[Reference 3, Table 11.1-7]

1. BAT Assessment Form – Tritium (cont.)	
Techniques to Prevent or Minimise at Source (options considered and selected)	Tritium derived from ternary fission is unavoidable in systems using uranium as a fuel. The ability to completely retain tritium within fuel pins is also considered infeasible since it diffuses readily through all suitable cladding materials. The use of zirconium cladding (ZIRLO® is used in the AP1000 NPP) reduces diffusion of tritium in comparison with other material options. The gray rod cluster assemblies are used in load following manoeuvring. The assemblies provide a mechanical shim reactivity mechanism to minimise the need for changes to the concentration of soluble boron. Burnable poisons are also employed in the initial cycle to limit the amount of boron required. [Reference 3, Section 4.1]. These measures minimise the production of tritium. Note that the use of enriched boron (B-10) does not reduce the amount of tritium produced since the source of tritium is predominantly from this isotope. However, it does reduce the total amount of boron required for chemical shim purposes. In the AP1000 NPP design it is considered that the use of gray rods eliminates the need for boron recycle and/or the use of more expensive enriched boron since boron usage is significantly reduced. Boron occurs naturally and is, therefore, relatively benign when discharged into the environment at design concentrations. The AP1000 NPP uses lithium hydroxide monohydrate enriched in the lithium-7 isotope to 99.9% for pH control (rather than Li6OH). This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/nickel-chromium-iron systems. In addition, lithium-7 is produced in solution from the neutron irradiation of the dissolved boron in the coolant. [Reference 3, subsection 5.2.3.2.1] Use of this isotope reduces the total amount of tritium produced in the AP1000 NPP compared to those designs utilising lithium-6, since the neutron absorption cross-section of lithium-7 is five orders of magnitude smaller than that of lithium-6. Both boron and lithium concentrations reduce as the fuel cycle advances, and thus, the production of tritium from these sources is reduced over time.

1. BAT Assessment Form – Tritium (cont.)

Techniques for minimisation at source – Optioneering

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Technique for minimising production of tritium isotope

(Green indicates techniques employed in AP1000 NPP)

Criteria (Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)	🟩 Lithium-7 Hydroxide	🟩 Zirconium cladding	Enriched boron	🟩 Gray rods	Boron recycle	
Proven Technology	2	2	2	2	2	
Available Technology	2	2	2	2	2	
Effective Technology	2	2	0	2	2	
Ease of Use	2	2	2	2	-1	
Cost	1	1	1	1	-2	
Impact (Public Dose)	1	1	0	1	1	
Impact (Operator Dose)	1	1	0	1	-2	
Impact (Environmental)	1	1	0	1	1	
Generates Suitable Waste Form	1	2	2	2	2	
Secondary & Decommissioning Waste	2	2	2	2	-1	
Totals	15	16	11	16	4	

Notes on scoring of minimisation techniques

a) The use of lithium-7 rather than lithium-6 reduces the production of tritium.

b) The particular zirconium cladding (ZIRLO®) used in the AP1000 NPP does not affect the amount of tritium produced. However, it is more effective than other clad materials in reducing the diffusion of tritium produced in the fuel through the cladding tube wall and thus into the primary coolant.

c) The use of enriched boron-10 does not in itself affect the amount of tritium produced since it is this isotope which is both a good neutron absorber (desirable characteristic) and produces tritium (undesirable characteristic). Although the AP1000 NPP uses a natural boron composition (20% B-10, 80% B-11), the design uses other measures (such as gray rods and neutron poisons including gadolinium) which reduce the overall boron requirement. Hence, it is considered that the impact of using enriched boron in the AP1000 NPP is a neutral value.

1. BAT Assessment Form – Tritium (cont.)	
Notes on scoring of minimisation techniques	d) The use of gray rods significantly reduces the amount of coolant borne boron needed for reactivity control. Since these are used to aid load following, fewer changes in boron concentration are required. This reduces the amount of boron used and therefore the amount of tritium produced through coolant borne boron reactions. ("Relatively little boric acid is used during power operation since load follow is accomplished with gray rods and without changes in the RCS boron concentration." [Reference 3, subsection 9.3.6.3.3]) e) Boron recycle systems do reduce the amount of tritium discharged to the environment. However, they also require large storage tanks (since recycle can only be carried out at the end of a cycle) and increased plant complexity (since more pumps and instrumentation are needed). This latter point is against the AP1000 NPP design philosophy of plant simplification and, therefore, among other aspects, reduced decommissioning and disposal needs at end of plant life. There is also an impact on operator dose. It is, therefore, considered that these additional cost and dose factors outweigh the environmental benefits gained from boron recycle.
Pathway to Environment from Source	Tritium commonly enters the environment as a replacement for one or more of the hydrogen atoms in water (HTO or T_2O), known as tritiated water, or in gaseous form (T_2 or HT) **Liquid Pathway** Around 1000 m^3 of tritiated reactor coolant is discharged (after processing) each year [Reference 3, Table 11.2-1], accounting for the majority of tritium discharges and the remainder being discharged to atmosphere or becoming incorporated in solid waste. This can be seen by comparing gaseous and liquid discharges of tritium. **Gaseous Pathway** The gaseous discharge route is described in Figures 11.3-1 and 11.3-2 of the AP1000 NPP Design Control Document [Reference 3]. Gaseous tritium is discharged to the atmosphere via the main ventilation stack.
Downstream Abatement Techniques (options considered and selected)	The principal measures for reducing the formation of tritium relate to the quality of the fuel cladding and the minimisation of fuel defects. A small proportion of the tritium (as gas) may be stripped out of the liquid effluent by the WLS degasifier and transferred to the gaseous radwaste system. However, the use of a condenser in this circuit ensures that gaseous tritiated water is returned to the liquid stream. This is beneficial since the potential dose is greater from the equivalent amount of gaseous tritium. Neither the liquid nor gaseous radwaste systems are able to remove tritium from the effluent or emission streams. Maintaining normal plant operations and minimising the need for unplanned plant shutdowns and depressurisation of the reactor cooling system is important for minimising tritium releases.

1. BAT Assessment Form – Tritium (cont.)

Downstream Abatement Techniques (cont)

(options considered and selected)

The liquid radwaste system comprising filtration and ion exchange beds is not effective in removing tritium.

The gaseous radwaste system comprising filtration and activated carbon delay beds is also ineffective at reducing tritium emissions because of the relatively long half-life of tritium.

Abatement Technique Optioneering (Tritium in Liquid)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For Tritium in Liquid** (Green indicates techniques employed in AP1000 NPP)								
	Adsorption	**Wet Scrubbing**	🟩 **Direct Discharge**	**Evaporators**	**Precipitation/ Filtration**	**Ion Exchange**	**Isotopic Concentration /Separation**	**Delay Tank**	🟩 **Minimise plant shutdowns**
Proven Technology	-2	-2	2	-2	-2	-2	2	-2	2
Available Technology	-2	-2	2	-2	-2	-2	2	-2	2
Effective Technology	-2	-2	2	-2	-2	-2	0	-2	2
Ease of Use	-2	-2	2	-2	-2	-2	-2	-2	1
Cost	0	0	2	0	0	0	-2	-2	0
Impact (Public Dose)	0	0	-1	0	0	0	1	0	1
Impact (Operator Dose)	0	0	1	0	0	0	-1	0	1
Impact (Environmental)	0	0	-1	0	0	0	1	0	1
Generates Suitable Waste Form	0	0	0	0	0	0	1	0	0
Secondary & Decommissioning Waste	-1	-1	0	-2	-1	-1	-1	-1	0
Totals	-9	-9	9	-10	-9	-9	1	-11	10

Notes on (Liquid) Abatement Techniques

a) Adsorption – no known application to tritium

b) Wet Scrubbing – applicable to particulate wastes but not to tritiated water

1. BAT Assessment Form – Tritium (cont.)	
Notes on (Liquid) Abatement Techniques (cont)	c) Direct discharge currently considered best option for tritiated water in absence of alternative technical viable or economically practicable alternative d) Evaporation – no benefit realised from this method since tritiated water behaviour is identical to non-tritiated water, and so no separation is achieved. An undesirable feature of evaporation is that tritiated water vapour could be discharged as part of the gaseous waste stream. e) Precipitation/filtration – no known techniques for removal of tritiated water from non-tritiated water using this method f) Ion Exchange – since tritiated water is chemically identical to non-tritiated water ion exchange methods cannot be used to discriminate between these different isotopic compositions. g) Isotopic concentration and/or separation is a possible method. However, it is considered that the investment required to develop the technology, and the costs involved in its implementation are significant in comparison to the very small component of tritium in the overall waste volume. h) The use of delay tanks for tritium is considered impractical because of its half life (12.4 years). The total volume of coolant would require storage for around 124 years to allow decay of tritium to background levels; thus, the tank volume would be unfeasibly large. i) Plant operation can significantly affect the amount of tritium produced. Therefore, good plant design, optimising plant availability, good training of operators, and the like are relevant contributors to minimisation of tritium production.

1. BAT Assessment Form – Tritium (cont.)

Abatement Technique Optioneering (Gaseous Tritium)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For Tritium in Liquid** (Green indicates techniques employed in AP1000 NPP)						
	Adsorption	**Direct Discharge**	**Isotopic Concentration /Separation**	**Carbon Delay Beds**	**Condenser**	**Cryogenics**	**Minimise plant shutdowns**
Proven Technology	-2	2	2	-2	2	2	2
Available Technology	-2	2	2	2	2	2	2
Effective Technology	-2	2	0	-2	2	2	2
Ease of Use	-2	2	-2	-2	1	-2	1
Cost	0	2	-2	-2	1	-2	0
Impact (Public Dose)	0	-1	1	1	1	1	1
Impact (Operator Dose)	0	1	-1	-2	0	-2	1
Impact (Environmental)	0	-1	1	0	1	1	1
Generates Suitable Waste Form	0	0	1	0	0	0	0
Secondary & Decommissioning Waste	-1	0	-1	-1	0	-1	0
Totals	-9	9	1	-8	10	1	10

a) Adsorption is not a process which can be applied to the separation of tritiated and non-tritiated water.

b) Direct discharge is currently considered best option for gaseous tritium (either T2 or HT) in absence of alternative technical viable or economically practicable alternative.

c) Evaporation – no benefit realised from this method since tritiated water/steam behaviour is identical to non-tritiated water/steam and so no separation is achieved. An undesirable feature of evaporation is that tritiated water vapour could be discharged as part of the gaseous waste stream.

d) Isotopic concentration and/or separation is a possible method. However, it is considered that the investment required developing the technology, and the costs involved in its implementation are significant in comparison to the very small component of tritium in the overall waste volume.

1. BAT Assessment Form – Tritium (cont.)	
Notes on (Gaseous) Abatement Techniques (cont)	e) Carbon Delay Beds – The use of delay beds for tritium is considered impractical because of its half life (12.4 years). The total gaseous waste arsing a delay period of around 124 years to allow decay of tritium to background levels. Storage of tritium gas is problematic since it diffuses through many commonly used structural and containment materials. The tanks would be unfeasibly large to accommodate the amount of tritium bearing gas for the delay period required. f) Condenser – Although this does not affect the levels of tritium discharged, it does affect the phase state in which it is discharged. The degassing system includes a gas cooler heat exchanger for removal of moisture entrained in the gas stream which is then routed to the WLS. This moisture (which includes tritiated water) would otherwise be discharged as part of the gaseous waste stream. Environmental benefits are claimed since tritium dose rates in water are less than air. g) Cryogenic systems could be used to liquefy tritium as part of a separation process. However, it is expensive both in terms of capital and operational costs. The use of complex equipment will result in higher operator dose, and it is likely to result in increased wastes. Long-term storage of separated tritium remains problematic because of its tendency to diffuse through many containment materials. h) Plant operation can significantly affect the amount of tritium produced. Therefore, good plant design, optimising plant availability, good training of operators, and the like are relevant contributors to minimisation of tritium production.
RP Predicted Normal Emission (TBq/y) (including allowance for normal operational fluctuation)	The expected tritium release from an AP1000 NPP is: 1.8 TBq/y to the atmosphere [Reference 5, 4.5, p. 12 and 5.1.2.3, Table 5-16] 33.4 TBq/y as liquid effluent [Reference 5, 4.5, p.10 and 5.1.1.3, Table 5-3]
RP Predicted Maximum Emission (TBq/y) (including allowance for normal operational fluctuation)	Worst Case Plant Discharge = 3.1 TBq/y to the atmosphere [Reference 6, Table 6.1-5]
Comparison with Emissions from Other Nuclear Power Stations	**South Texas 1** 4.181 TBq/y gaseous 58.1 TBq/y liquid [Reference 5, 5.1.3, p. 33, Table 5-13]

<table>
<tr><th colspan="2">1. BAT Assessment Form – Tritium (cont.)</th></tr>
<tr><td>Comparison with Emissions from Other Nuclear Power Stations (cont)</td><td>
Braidwood 1
0.64 TBq/y gaseous
58.1 TBq/y liquid
[Reference 5, 5.1.3, p. 34, Table 5-14]
Cook 1
5.0 TBq/y gaseous
45.88 TBq/y liquid
[Reference 5, 5.1.3, p. 35, Table 5-15]
Vogtle 1
48.2 TBq/y gaseous
47.36 TBq/y liquid
[Reference 5, 5.1.3, p. 36, Table 5-16]
Sizewell B - Predicted
3 TBq/y gaseous
60 TBq/y liquid
[Reference 5, 5.1.3, p. 37, Table 5-17, and Reference 2]
Comparison of AP1000 NPP Liquid Radioactive Discharges of Tritium with European Nuclear Power Plants Between 1995 and 1998
<table>
<tr><th></th><th>Unit</th><th>AP1000 NPP</th><th>Sizewell B</th><th>All PWR</th><th>All Magnox and AGR</th><th>All BWR</th></tr>
<tr><td>No Plants</td><td></td><td>0</td><td>1</td><td>73</td><td>30</td><td>10</td></tr>
<tr><td>Minimum</td><td>TBq/ GWa</td><td>30.5</td><td>17.9</td><td>0.02</td><td>88</td><td>0.34</td></tr>
<tr><td>Average</td><td>TBq/ GWa</td><td>33.4</td><td>36.1</td><td>16.2</td><td>357.15</td><td>0.83</td></tr>
<tr><td>Maximum</td><td>TBq/ GWa</td><td>35.1</td><td>45.9</td><td>45.9</td><td>463</td><td>1.92</td></tr>
</table>
[Reference 6, Table 3.4-19]
</td></tr>
</table>

1. BAT Assessment Form – Tritium (cont.)	
Associated H-3 Dose (µSv/y)	Marine discharge – fisherman family dose 2.4E-02 µSv/y [Reference 6, Table 5.2.12] Aerial discharge – local resident family dose 8.6E-02 µSv/y [Reference 6, Table 5.2.16]
H-3 Dose as a Percentage of Total Dose	Marine discharge – fisherman family dose ~1% Aerial discharge – local resident family dose ~2.5%

References

1. "Radionuclides Handbook," R&D Technical Report P3-101/SP1b, Environment Agency, October 2003.
2. "Review of the Control and Impact of the Discharge and Disposal of Radioactive Waste at Sizewell B Power Station," SZB/THR/042, Revision 000, British Energy Generation Ltd.
3. EPS-GW-GL-700, Rev 1, "AP1000 European Design Control Document," Westinghouse Electric Company LLC, 2009.
4. "Nuclear Engineering – Theory and Technology of Commercial Nuclear Power," Ronald Allen Knief, Taylor and Francis, 1992.
5. APP-WLS-M3C-049, Revision 2 "Monthly Radiation Emissions from Radioactive Nuclides," AP1000 Calculation Note (proprietary) Westinghouse Electric Company LLC, 2009.
6. UKP-GW-GL-790, Rev 4, "UK AP1000 Environment Report," Westinghouse Electric Company LLC, 2011.

2. BAT Assessment Form – C-14

Individual Radionuclide Information			
Radionuclide	C-14	**Discharge Route (Solid/Liquid/Gas)**	Gas/Liquid
Radioactive Half-life	5730 years	**Parent** **Daughter**	N/A N-14
Principal Decay Mode	Beta	**Average Energy**	0.0495 MeV
Speciation		The majority of compounds of carbon are in the +4 oxidation state. Its chemistry is characterised by its tendency to form stable bonds with oxygen, hydrogen, halides, nitrogen, sulphur and other carbon atoms. In solution, the carbonate and bicarbonate ions predominate. [Reference 1, p. 40] Carbon may be added to the RCS as natural C-12 by the following mechanisms: • dissolved carbon dioxide in the RCS makeup water (~200g carbon per cycle), • by total organic carbon (TOC) contamination within the demineralised water (~50g carbon per cycle, assuming the 50ppb specification for TOC in demineralised water) • by the addition of zinc acetate (see Section 4.1.3.13) for corrosion control (~1-4kg carbon per cycle). With the large excess of hydrogen in the reactor coolant during operation, the resultant products of the radiolysis of acetate are likely to be fully reduced species: methane, ethane, and the carbon that has been detected in core crud While isotopic exchange between the C-12 and the C-14 generated will occur in the radiation field of the reactor core, it is not expected that this will change the partitioning of C-14 between the liquid and gas phases. The majority of the C-12 introduced end up in the gas phase mainly in the form of methane.	
Source of Radioactivity (back to the point of generation)		In a PWR reactor Carbon-14 may be produced by: • Neutron activation of oxygen in the water coolant (O-17 (n, α) → C-14); • Neutron activation of nitrogen in the water coolant (N-14 (n,p) → C-14). [Reference 2, p. 23] Airborne release of C-14 from PWRs is predominantly hydrocarbons (75 - 95%), mainly methane, with only a small fraction in the form of CO_2. [Reference 4, p. 14] The production of C-14 is estimated to be 6g per cycle, which is several orders of magnitude less than the C-12 added to the RCS by the mechanisms described above.	

BAT Assessment Form – C-14 (cont.)	
Source Activity **(before abatement)**	**Design Basis [Reference 9]** C-14 production from O-17 (n, α): 552 GBq/y C-14 production from N-14 (n,p): 110 GBq/y* Total C-14 production 662 GBq/y *Based on a dissolved nitrogen concentration of 15ppm in the primary coolant which is the equilibrium concentration for water saturated with nitrogen from air at atmospheric conditions
Techniques to Prevent or Minimise at Source (options considered and selected)	According to the IAEA [Reference 4, p. 37] around 0.15 TBq per GWa of C-14 is produced by the activation of O-17 in LWR fuel. This represents an irreducible minimum since the amount of oxygen present in the fuel is somewhat inflexible to adjustment. However, the nitrogen impurities in the fuel may be controlled during fuel fabrication. The AP1000 NPP uses fuel rods pre-pressurised with helium to minimise compressive clad stresses and prevent clad flattening under reactor coolant operating pressures. The use of helium pressurisation expels nitrogen from the fuel rod. Since water is used as a coolant and moderator, the production of C-14 by means of oxygen activation is unavoidable in PWR systems. The use of another coolant medium is not a viable option as the use of light water is fundamental to this technology. Control of water chemistry by oxygen scavenging, pH control using Li_7OH and electrodeionisation comprise the primary process used on the AP1000 NPP to reduce C-14 formation. Following refuelling, the RCS is refilled with boric acid made from demineralised water produced by the demineralised water treatment system (DTS). The DTS comprises cartridge filtration, two reverse osmosis units and an electrodeionisation unit. Electrodeionisation is used for secondary demineralization and the removal of dissolved carbon dioxide gas. [Reference 3, subsection 9.2.3.2.2] Oxygen control of the demineralised water is performed by catalytic oxygen reduction units which reduce oxygen levels to <100ppb or less. [Reference 3, subsection 9.2.4.1.2] Two catalytic oxygen reduction units are used in the AP1000 NPP plant. One unit is provided for the demineralised water distribution system, and the second is provided at the condensate storage tank to maintain low oxygen content within the tank. [Reference 3, subsection 9.2.4.2.2] Each catalytic oxygen reduction unit consists of a mixing chamber, a catalytic resin vessel, and a resin trap. Dissolved oxygen is removed chemically by mixing the effluent from the storage tank with hydrogen gas supplied from the plant gas system. The AP1000 NPP also reduces oxygen (and hence C-14 production) by means of the CVS system. The CVS maintains the proper conditions in the RCS to minimise corrosion of the fuel and primary surfaces. During power operations, dissolved hydrogen is added to the RCS to eliminate free oxygen (produced by radiolysis in the core) and to prevent ammonia formation. During plant start up from cold shutdown, the CVS introduces an oxygen scavenger into the RCS. This system is only used for oxygen control at low reactor temperatures during start-up from cold shutdown conditions. [Reference 3, subsection 9.3.6.1.2.4, 9.3.6.2.4.1]

2. BAT Assessment Form – C-14 (cont.)

In the AP1000 NPP vacuum fill is used on the reactor coolant system as part of start-up after refueling. While this procedure very substantially reduces air which is trapped in the coolant at startup, the relatively large volume of the AP1000 NPP steam generator tubes means that substantial mass of air is trapped, even under these evacuated conditions. While the oxygen is then removed by chemical means, the nitrogen will remain in solution. Similarly, the boric acid tank is assumed to be saturated with air. In reality, since the major use of this tank will be in preparation for shutdown, the ultimate contribution to C14 from this source will be small.

According to the IAEA [Reference 4], the cover gas of water storage tanks can be switched from nitrogen to argon so as to prevent the dissolution of nitrogen in coolant, thereby reducing C-14 production. The AP1000 NPP does not use any techniques for removing dissolved nitrogen from the demineralised water and the water tanks in the AP1000 NPP are not blanketed with either argon or nitrogen. Instead, catalytic oxygen reduction units are used to reduce dissolved oxygen levels through recirculation and immediately before the water is pumped to the distribution header.

pH control of the primary coolant using lithium hydroxide instead of hydrazine (NH2-NH2) prevents formation of C-14 from nitrogen and is used in the AP1000 NPP.

Techniques for minimisation at source – Optioneering

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Technique for minimising production of tritium isotope** (Green indicates techniques employed in AP1000 NPP)				
	🟩 **Oxygen scavenging**	🟩 **Control of nitrogen impurities in fuel**	**Argon v nitrogen cover gas**	🟩 **pH control by lithium hydroxide**	🟩 **Electro-deionisation**
Proven Technology	2	2	2	2	2
Available Technology	2	2	2	2	2
Effective Technology	2	1	1	2	2
Ease of Use	2	2	-1	2	2
Cost	0	0	-1	0	-1
Impact (Public Dose)	1	1	1	1	1
Impact (Operator Dose)	1	1	1	1	1
Impact (Environmental)	1	1	1	1	1
Generates Suitable Waste Form	0	0	0	0	0
Secondary & Decommissioning Waste	0	0	0	0	0
Totals	11	10	6	11	10

2. BAT Assessment Form – C-14 (cont.)	
Notes on scoring of minimisation techniques	
Pathway to Environment from Source	The substrate atoms for the activation reactions occur widely in fuel, cladding, coolant or structural materials either as major constituents or as impurities. In consequence, C-14 produced in a nuclear power reactor can be released directly to the environment from the coolant in a gaseous form or as liquid effluent. [Reference 4, p. 12]. The majority of the C-14 is released in gaseous form via the main plant vent.
Pathway to Environment from Source (cont.)	C-14 is produced in the fuel and in stainless steel structural materials. It can be assumed that C-14 will remain within these materials and will either be a constituent part of decommissioning wastes (for structural materials) contained as a solid waste (in clad material) or, if fuel is reprocessed, passed to the reprocessing facility. In the latter case, C-14 will be released to the off-gas system during dissolution. [Reference 4]
Downstream Abatement Techniques (options considered and selected)	**Liquid Abatement Techniques** • Ion Exchange • Evaporation **Gaseous Abatement Techniques – Carbon Dioxide** • Alkaline slurry scrubber • Alkaline packed bed column • Double alkali process • Gas absorption by wet scrubbing • Ethanolamine scrubbing • Absorption in a fluorocarbon solvent • Physical absorption on an active surface • Reaction with magnesium • Isotopic separation Note 1 – Details of these methods can be found in Reference 4 with the exception of evaporation. Note 2 – The majority of emissions from PWRs generally (and from the AP1000 NPP) are in the form of hydrocarbons, predominantly methane. For physical and chemical reasons, CO_2 is the carbon compound that can most easily be separated from other gases. The treatment processes focus on the removal of CO_2 from gas streams and, therefore, assumes C-14 compounds are oxidised to $^{14}CO_2$ before removal by, for example, high temperature catalytic oxidation. These steps in themselves represent significant technical challenges, and thus, the assessment markings are adjusted to make allowance for these difficulties.

2. BAT Assessment Form – C-14 (cont.)

<table>
<tr>
<td>Downstream Abatement Techniques (cont)
(options considered and selected)</td>
<td><u>Note 3</u> – According to the OSPAR Commission [Reference 5, p. 20], in their national reports, Contracting Parties generally acknowledged that operational management systems are in place to prevent, eliminate, or reduce liquid waste. Such systems are an essential element of the application of BAT. In addition, the abatement techniques identified in the NEA and IAEA reports on available liquid effluent options have been employed by Contracting Parties individually or in combination, to remove particular materials and nuclides (except tritium and carbon 14) from the liquid effluents. For tritium and carbon 14 (two nuclides which require particular attention according to Bremen agreement), implementation reports of the Contracting Parties do not mention that any abatement technique has been implemented for tritium and carbon 14 in the liquid effluents discharged by the nuclear industry. There is a significant level of agreement in the processes being employed, which provides a strong indication that international best practice – and by extension BAT – is being applied.</td>
</tr>
<tr>
<td>Abatement Technique Optioneering (C-14 in Liquid)
(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)</td>
<td>
<table>
<tr><td rowspan="2">Criteria
(Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)</td><td colspan="8">Abatement Technique For C-14 in Liquid
(Green indicates techniques employed in AP1000 NPP)</td></tr>
<tr><td>🟩 Ion Exchange</td><td>🟩 Direct discharge</td><td>Evaporation</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Proven Technology</td><td>2</td><td>2</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Available Technology</td><td>2</td><td>2</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Effective Technology</td><td>0</td><td>0</td><td>-2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Ease of Use</td><td>2</td><td>2</td><td>-2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Cost</td><td>-1</td><td>2</td><td>-2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Public Dose)</td><td>0</td><td>-1</td><td>0</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Operator Dose)</td><td>-1</td><td>2</td><td>-2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Environmental)</td><td>0</td><td>-1</td><td>0</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Generates Suitable Waste Form</td><td>1</td><td>0</td><td>-1</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Secondary & Decommissioning Waste</td><td>-2</td><td>2</td><td>-2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Totals</td><td>3</td><td>10</td><td>-7</td><td></td><td></td><td></td><td></td><td></td></tr>
</table>
</td>
</tr>
</table>

2. BAT Assessment Form – C-14 (cont.)

Notes on (Liquid) Abatement Techniques	Ion Exchange in the form of cation and mixed resin beds is used in the AP1000 NPP as the primary abatement technique for removal of trace metal radionuclides (for example, Co-60 and Cs-137). C-14 in the form of carbonate and bicarbonate is removed by the mixed resin bed. Evaporation – No benefit realised from this method since C-14 is largely produced in the gas phase.
Abatement Technique Optioneering (Gaseous C-14) (Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)	

Criteria (Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For C-14 in Gas** (Green indicates techniques employed in AP1000 NPP)					
	Alkaline slurry scrubber	**Alkaline packed bed column**	**Double alkali process**	**Gas absorption by wet scrubbing**	**Ethanolamine scrubbing**	**Absorption in a fluorocarbon solvent**
Proven Technology	2	2	-1	1	1	1
Available Technology	1	1	0	1	1	1
Effective Technology	2	2	2	1	1	2
Ease of Use	-1	-1	-1	-1	-2	-1
Cost	-1	-1	-1	-2	-2	-2
Impact (Public Dose)	2	2	2	2	2	2
Impact (Operator Dose)	-1	-1	-1	-1	-1	-1
Impact (Environmental)	2	2	2	2	2	2
Generates Suitable Waste Form	-2	-2	-2	-2	-2	-2
Secondary & Decommissioning Waste	-1	-1	-1	-1	-1	-2
Sub Totals	3	3	-1	0	-1	0
Allowance to compensate for conversion of hydrocarbon to CO_2	-2	-2	-2	-2	-2	-2
Total	1	1	-3	-2	-3	-2

2. BAT Assessment Form – C-14 (cont.)

Abatement Technique Optioneering (Gaseous C-14)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For C-14 in Gas (cont)** (Green indicates techniques employed in AP1000 NPP)					
	Physical absorption on an active surface	Reaction with magnesium	Isotopic separation	Cryogenics	Direct discharge	
Proven Technology	0	-1	1	1	2	
Available Technology	1	0	-1	-1	2	
Effective Technology	2	2	1	1	0	
Ease of Use	-2	-1	-2	-2	2	
Cost	-2	-2	-2	-2	2	
Impact (Public Dose)	2	2	1	1	-2	
Impact (Operator Dose)	-1	-1	-1	-1	2	
Impact (Environmental)	2	2	1	1	-2	
Generates Suitable Waste Form	-1	-2	0	0	0	
Secondary & Decommissioning Waste	-1	-1	-1	-1	2	
Totals	0	-2	-3	-3	8	
Allowance to compensate for conversion of hydrocarbon to CO_2	-2	-2	-2	-2	0	
Total	-2	-4	-5	-5	8	

Notes on (Gaseous) Abatement Techniques

Alkaline slurry scrubber and alkaline packed bed technologies have both been studied extensively, but are only presently at a stage where demonstration plant work can commence. Suitability of secondary wastes for subsequent treatment unknown. Additional plant will result in additional decommissioning wastes.

2. BAT Assessment Form – C-14 (cont.)	
Notes on (Gaseous) Abatement Techniques (cont.)	Double alkali process. Process control is critical in this technology to avoid plugging of the packed column. It is currently not known whether this technology has been trialled for nuclear use. Gas absorption by wet scrubbing. Studies have indicated that a large column is needed to attain near complete removal of CO_2. Space limitations in a nuclear plant make this method less feasible than other methods. Ethanolamine scrubbing. A product solidification technique would be needed in addition to this process. Oxidation of ethanolamine to corrosive oxalic acid and glycine present technical issues. Absorption in a fluorocarbon solvent. Demonstrated for Kr-85 but unproven for CO_2. Contamination of CO_2 product with Kr-85 may complicate disposal. Physical absorption on an active surface. This process requires pre-drying of gas stream and post-processing utilising the double alkali process. Bed temperature is critical parameter during operation. Isotopic concentration and/or separation is a possible method. However, it is considered that the investment required developing the technology and the costs involved in its implementation are significant in comparison to the very small component of tritium in the overall waste volume. Cryogenic systems could be used to liquefy CO_2 as part of a separation process. However, it is expensive both in terms of capital and operational costs. The use of complex equipment will result in higher operator dose, and it is likely to result in increased wastes.
RP Predicted Emission (including allowance for normal operational fluctuation)	The expected Carbon-14 release from an AP1000 NPP is [Reference 9]: Solid Waste 53.0 GBq/y Liquid Release 3.3 GBq/y Gaseous Release 606 GBq/y
Comparison with Emissions from Other Nuclear Power Stations	**Sizewell B – Predicted** The predicted future rolling 12-month discharge of Carbon-14 is 300-500 GBq to the atmosphere. [Reference 2, p. 79]

2. BAT Assessment Form – C-14 (cont.)

Comparison with Emissions from Other Nuclear Power Stations (cont.)

Mean and standard deviation of the data available for predecessor designs (Airborne carbon-14) [Reference 7, Table 2]

Design	Mean GBq/GWeh	Standard GBq/GWeh deviation	Maximum GBq/GWeh	Predicted GBq/GWeh
AP1000 NPP	1.80E-02	8.6E-03	2.66E-02	2.76E-02
EPR	3.07E-02	1.39E-02	4.46E-02	2.41E-02
ESBWR	n/av	n/av	n/av	n/av
ACR1000	1.81E-01	2.36E-01	4.17E-01	2.95E-02

Normalised releases of radionuclides from nuclear reactors (TBq/GWy) [Reference 8 Table 37]

Year	PWR	BWR	HWR
1970-1974	0.22	0.52	6.3
1975-1979	0.22	0.52	6.3
1980-1984	0.35	0.33	6.3
1985-1989	0.12	0.45	4.8
1990-1994	0.22	0.51	1.6

Associated C-14 Dose (µSv/y)

Marine discharge – fisherman family dose 1.6E+00 µSv/y

[Reference 6, Table 5.2.12]

Aerial discharge – local resident family dose 3.3E+00 µSv/y

[Reference 6, Table 5.2.16]

C-14 Dose as a Percentage of Total Dose

Marine discharge – fisherman family dose ~70%

Aerial discharge – local resident family dose ~92%

2. BAT Assessment Form – C-14 (cont.)

References

1. "Radionuclides Handbook." R&D Technical Report P3-101/SP1b, Environment Agency, October 2003.
2. "Review of the Control and Impact of the Discharge and Disposal of Radioactive Waste at Sizewell B Power Station," SZB/THR/042, Revision 000, British Energy Generation Ltd.
3. EPS-GW-GL-700, Rev 1 "AP1000 European Design Control Document," Westinghouse Electric Company LLC, 2009.
4. Management of Waste Containing Tritium and Carbon-14, Technical Reports Series Number 421, International Atomic Energy Agency, July 2004.
5. "Assessment of the 4th Round of Reporting on the Implementation of PARCOM Recommendations 91/4 on Radioactive Discharges," OSPAR Commission, ISBN 978-1-905859-90-0.
6. UKP-GW-GL-790, Rev 4, "UK AP1000 Environment Report," Westinghouse Electric Company LLC, 2011.
7. "Study of historic nuclear reactor discharge data," Coppleston, D et al., Radioprotection, Vol 44, No 5, (2009) 875 – 880.
8. UNSCEAR 2000 Report, Vol 1, Annex C. United Nations Scientific Committee on the Effects of Atomic Radiation.
9. APP-WLS-M3C-056, Rev 0, "AP1000 Expected Production and Release Rates for 14C," Westinghouse Electric Company LLC, 2009.

<table>
<tr><th colspan="4">3. BAT Assessment Form – N-16</th></tr>
<tr><td colspan="4">Individual Radionuclide Information</td></tr>
<tr><td>Radionuclide</td><td>N-16</td><td>Discharge Route (Solid/Liquid/Gas)</td><td>Gas/Liquid</td></tr>
<tr><td rowspan="2">Radioactive Half-life</td><td rowspan="2">7.13s</td><td>Parent</td><td>O-16</td></tr>
<tr><td>Daughter</td><td>O-16</td></tr>
<tr><td>Principal Decay Mode</td><td>Beta</td><td>Average Energy</td><td>10.42 MeV</td></tr>
<tr><td colspan="2">Speciation</td><td colspan="2">Nitrogen in solution tends to be found in anionic forms nitrite (NO^{2-}) or nitrate (NO^{3-}) or cationic form ammonium (NH^{4+}). Gaseous forms of nitrogen include nitrogen gas (N_2), ammonia (NH_3) or nitrogen oxides (N_2O, NO or NO_2). A variety of organic compounds may also contain nitrogen.</td></tr>
<tr><td colspan="2">Source of Radioactivity
(back to the point of generation)</td><td colspan="2">The activation of oxygen in the primary coolant results in the formation of N-16 which is a strong gamma emitter. Due to its short half life of 7.1 seconds, N-16 is not a concern outside the containment. N-16 is the predominant contributor to the activity in the reactor coolant pumps, steam generators and reactor piping during operation. The activity in each component depends on the transit time to the component and the residence time.
The secondary shield surrounding the RCS equipment (including piping, pumps and steam generators) protects personnel in power operation.
The source is terminated when the reactor shuts down.
[Reference 3, Table 13.3-1, p. 452, and Reference 3 subsection 11.1.1.4]</td></tr>
<tr><td colspan="2">Source Activity (Bq)
(before abatement)</td><td colspan="2">N/A (The source is terminated when the reactor shuts down).</td></tr>
<tr><td colspan="2">Techniques to Prevent or Minimise at Source
(options considered and selected)</td><td colspan="2">The RCS is filled with water and, therefore, oxygen atoms. Below only removes molecular oxygen. An important mechanism for minimising the formation of nitrogen-16 is the reduction of oxygen levels in the coolant.
High concentrations of oxygen in the RCS will lead to increased production of nitrogen-16. Degassing of the demineralised water and the detection of oxygen ingress to the demineralised water are important to minimise nitrogen-16 production.
Following refueling, the RCS is refilled with boric acid made from demineralised water produced by the DTS. The DTS comprises cartridge filtration, two reverse osmosis units and an electrode ionisation unit. Electrode ionization is used for secondary demineralization and the removal of dissolved carbon dioxide gas.
Oxygen control of the demineralised water is performed by catalytic oxygen reduction units which reduce oxygen levels to <100ppb. Two catalytic oxygen reduction units are used in the AP1000 NPP. One unit is provided for the demineralised water distribution system, and the second is provided at the condensate storage tank to maintain low oxygen content within the tank. Each catalytic oxygen reduction unit consists of a mixing chamber, a catalytic resin vessel, and a resin trap. Dissolved oxygen is removed chemically by mixing the effluent from the storage tank with hydrogen gas supplied from the plant gas system.</td></tr>
</table>

3. BAT Assessment Form – N-16 (cont.)

Techniques to Prevent or Minimise at Source (cont.) (options considered and selected)	In addition to oxygen control in the demineralised water system, the CVS provides control of the RCS oxygen concentration both during startup by introduction of hydrazine and during power operations by injection of hydrogen. The latter drives the equilibrium concentration of oxygen produced by radiolysis in the core toward zero.

Techniques for minimisation at source – Optioneering

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Technique for minimisation production of N-16 isotope** (Green indicates techniques employed in AP1000 NPP)					
	Hydrazine Addition	**Oxygen Elimination**				
Proven Technology	2	2				
Available Technology	2	2				
Effective Technology	2	2				
Ease of Use	1	1				
Cost	-1	-1				
Impact (Public Dose)	0	0				
Impact (Operator Dose)	2	2				
Impact (Environmental)	2	2				
Generates Suitable Waste Form	2	2				
Secondary & Decommissioning Waste	2	2				
Totals	14	14				

Notes on scoring of minimisation techniques	None
Pathway to Environment from Source	Nitrogen-16 predominantly decays within containment before it can be released through the gaseous or liquid radwaste systems. Nitrogen-16 detectors that are sensitive for detecting primary-to-secondary coolant leakage are located near the steam generator main steam outlet and upstream of the turbine. Nitrogen-16 is not a major contributor to gaseous or liquid releases of radioactivity to the environment.
Downstream Abatement Techniques (options considered and selected)	The nitrogen-16 activity is not a factor in the radiation sources for systems and components located outside containment. This is due to its short half-life (7.13 seconds) and the greater than one minute transport time before flow exits the containment. The activated carbon delay beds in the gaseous radwaste system provide a delay time that, in conjunction with the short half-life (7.13 seconds), allows further reduction of Nitrogen-16 activity before discharge to the environment.

<table>
<tr><th colspan="2">3. BAT Assessment Form – N-16 (cont.)</th></tr>
<tr><td>Downstream Abatement Techniques (cont.)
(options considered and selected)</td><td>Nitrogen-16 discharges in liquid effluent will be either in cationic of anionic form and will be absorbed or retarded by the ion exchange beds in the liquid radwaste treatment system. The result is that nitrogen-16 is not a major contributor to the activity or dose rates associated with liquid effluent discharges.</td></tr>
<tr><td>Abatement Technique Optioneering (N-16)
(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)</td><td>
<table>
<tr><th rowspan="2">Criteria
(Scoring -2 to 2 – with 2 good, -2 poor, 0 indicates neither a benefit nor a disadvantage)</th><th colspan="9">Abatement Technique For N-16
(Green indicates techniques employed in AP1000 NPP)</th></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Proven Technology</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Available Technology</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Effective Technology</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Ease of Use</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Cost</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Public Dose)</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Operator Dose)</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Environmental)</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Generates Suitable Waste Form</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Secondary & Decommissioning Waste</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Totals</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>
</td></tr>
<tr><td>Notes on (Liquid) Abatement Techniques</td><td>No abatement processes are necessary since N-16 rapidly decays before it can be released to the environment.</td></tr>
<tr><td>RP Predicted Normal Emission (TBq/y)
(including allowance for normal operational fluctuation)</td><td>N-16 release is assumed to be negligible as it is found not to be released when running the GALE code. [Reference 2, p. 12]</td></tr>
</table>

3. BAT Assessment Form – N-16 (cont.)	
Comparison with Emissions from Other Nuclear Power Stations	N-16 release is assumed to be negligible as it is found not to be released when running the GALE code. [Reference 2, p. 12]
Associated N-16 Dose (µSv/y)	N-16 release is assumed to be negligible as it is found not to be released when running the GALE code. [Reference 2, p. 12]
N-16 Dose as a Percentage of Total Dose	N-16 release is assumed to be negligible as it is found not to be released when running the GALE code. [Reference 2, p. 12]

References

1. EPS-GW-GL-700, Rev 1, “AP1000 European Design Control Document,” Westinghouse Electric Company LLC, 2009.
2. APP-WLS-M3C-049, Revision 2 “Monthly Radiation Emissions from Radioactive Nuclides,” AP1000 Calculation Note (proprietary / protect commercial) Westinghouse Electric Company LLC, 2009.
3. UKP-GW-GL-793, Rev 0 “AP1000 Pre-Construction Safety Report,” Westinghouse Electric Company LLC, 2011.

4. BAT Assessment Form – Sr-90

Individual Radionuclide Information			
Radionuclide	Sr-90	**Discharge Route (Solid/Liquid/Gas)**	Gas / Liquid
Radioactive Half-life	29.1 years	**Parent** **Daughter**	N/A Y-90[R]
Principal Decay Mode	Beta	**Average Energy**	0.196 MeV
Speciation		Strontium is an alkaline earth element and, thus, the most important species is the Sr^{2+} ion. Isotopes of strontium can be expected to take part in a number of precipitation and substitution reactions. Precipitation as sulphate, carbonate or hydroxide is possible. [Reference 1, p. 168]	
Source of Radioactivity (back to the point of generation)		It is believed that Strontium-90 occurs in the coolant primarily as the result of fission fragment recoil or as a result of the decay of short-lived noble gas precursors in water. [Reference 3]	
Source Activity (Bq) (before abatement)		**Design Basis** Reactor Coolant Activity: 4.9 E-05 µCi/g (1.813 Bq/g) based on the conservative assumption of 0.25% fuel defects [Reference 2, Table 11.1-2] Steam Generator Secondary Side Liquid Activity: 1.5E -07 µCi/g (0.00555 Bq/g) based on the conservative assumption of a primary to secondary leak rate of 500 gpd) [Reference 2, Table 11.1-5] **Realistic Source Terms** Reactor Coolant Activity: 1.0E-05 µCi/g (0.37 Bq/g) Steam Generator Liquid Activity: 1.4 E -9 µCi/g (5.18 E -05 Bq/g) Steam Generator Steam Activity: 7.0E-12 µCi/g (2.59 E-07 Bq/g) [Reference 2, Table 11.1-8] Water mass in RCS – 4.3 x 10^5 lbs [195.044 x 10^3 kg] Total secondary side water mass in Steam Generators – 3.5 x 10^5 lbs [158.757 x 10^3 kg] [Reference 2, Table 11.1-7]	

<table>
<tr><th colspan="2">4. BAT Assessment Form – Sr-90 (cont.)</th></tr>
<tr><td>Techniques to Prevent or Minimise at Source
(options considered and selected)</td><td>The production of strontium-90 is unavoidable in fission reactors using uranium as a fuel (see note on ‘Source of radioactivity’ above).
A proportion of strontium may be deposited on surfaces within the reactor system by mechanisms such as plateout – however no decontamination factor is applied for the removal of Sr-90 by the chemical and volume control system cation bed demineraliser.
[Reference 2, Notes to Table 11.1-1]</td></tr>
<tr><td>Techniques for minimisation at source – Optioneering
(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT)</td><td>
<table>
<tr><th rowspan="2">Criteria
(Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)</th><th colspan="6">Technique for minimising production of strontium-90 isotope
(Green indicates techniques employed in AP1000 NPP)</th></tr>
<tr><th>🟩 Minimisation of fuel defects in operation</th><th></th><th></th><th></th><th></th><th></th></tr>
<tr><td>Proven Technology</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Available Technology</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Effective Technology</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Ease of Use</td><td>1</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Cost</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Public Dose)</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Operator Dose)</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Environmental)</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Generates Suitable Waste Form</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Secondary & Decommissioning Waste</td><td>2</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Totals</td><td>19</td><td></td><td></td><td></td><td></td><td></td></tr>
</table>
</td></tr>
<tr><td>Notes on scoring of minimisation techniques</td><td>Scoring is not appropriate because the production of Strontium-90 is unavoidable in fission reactors using uranium as a fuel.</td></tr>
<tr><td>Pathway to Environment from Source</td><td>Strontium-90 is discharged to the environment both through liquid discharge and to atmosphere via the main ventilation stack after pre-treatment.</td></tr>
<tr><td>Downstream Abatement Techniques
(options considered and selected)</td><td>The AP1000 NPP is equipped with a mixed bed demineraliser which removes a proportion of strontium isotopes from liquid effluent.
The demineraliser has an isotopic decontamination factor of 10 for strontium isotopes [Reference 2, Table 11.1-1]
The WLS of the AP1000 NPP comprising filtration and ion exchange beds is effective in reducing strontium-90 levels to within acceptable discharge</td></tr>
</table>

4. BAT Assessment Form – Sr-90 (cont.)	
Downstream Abatement Techniques (cont.) (options considered and selected)	limits. Ion exchange in particular is recognised as a very effective treatment method for the removal of strontium. Although more effective ion exchange materials may be available for strontium removal, the choice made for the AP1000 NPP optimises the requirement for removal of certain radiological species with other chemical and operational constraints as outlined by the IAEA below. "Nuclear power plant process water systems have typically used organic ion exchange resins to control system chemistry to minimise corrosion or the degradation of system components and to remove radioactive contaminants. Organic resins are also used in a number of chemical decontamination or cleaning processes for the regeneration of process water by reagents and for radionuclide removal. In the past decade inorganic ion exchange materials have emerged as an increasingly important replacement or complement for conventional organic ion exchange resins, particularly in liquid radioactive waste treatment and spent fuel reprocessing applications. Inorganic ion exchangers often have the advantage of a much greater selectivity than organic resins for certain radio logically important species, such as caesium and strontium. These inorganic materials may also prove to have advantages with respect to immobilization and final disposal when compared with organic ion exchangers. However, in nuclear power plant operations the currently available inorganic exchangers cannot entirely replace conventional organic ion exchange resins, especially in high purity water applications or in operations in which the system chemistry must be controlled through the maintenance of dissolved species such as lithium ions or boric acid." [Reference 5, p. 1]

4. BAT Assessment Form – Sr-90 (cont.)

Abatement Technique Optioneering (Strontium in Liquid)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2 –with 2 good, -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For Strontium in Liquid** (Green indicates techniques employed in AP1000 NPP)							
	Ion Exchange	**Wet Scrubbing**	**Direct Discharge**	**Evaporators**	**Precipitation/ Filtration**	**Adsorption**	**Isotopic Concentration /Separation**	**Delay Tank**
Proven Technology	2	-2	2	2	1	-2	2	-2
Available Technology	2	-2	2	2	1	-2	2	-2
Effective Technology	2	-2	-2	2	1	-2	-1	-2
Ease of Use	2	-2	2	-2	1	-2	-2	-2
Cost	-1	-2	2	-2	-1	0	-2	-2
Impact (Public Dose)	2	0	-2	2	2	-2	1	2
Impact (Operator Dose)	-1	0	1	-2	-1	-1	-2	-1
Impact (Environmental)	2	0	-2	2	2	-2	1	2
Generates Suitable Waste Form	-1	0	0	-2	-1	0	-1	-1
Secondary & Decommissioning Waste	-1	-2	0	-2	-1	-1	-1	-2
Totals	8	-12	3	0	4	-14	-3	-10

Notes on (Liquid) Abatement Techniques

a) Ion Exchange – considered to be a highly effective and industry preferred method for strontium removal

b) Wet Scrubbing – no application for strontium removal

c) Direct discharge – environmentally not acceptable option

d) Evaporation – no benefit realised from this method. Anticipated to be highly expensive and technically challenging to isolate small amounts of strontium

e) Precipitation/filtration – not considered particularly effective for strontium

f) Adsorption – no known application for strontium

g) Isotopic concentration and/or separation is a possible method. However, it is considered that the investment required to develop the technology and the costs involved in its implementation are significant in comparison to the very small component of strontium in the overall waste volume.

4. BAT Assessment Form – Sr-90 (cont.)

Notes on (Liquid) Abatement Techniques (cont.)

The use of delay tanks for strontium is considered impractical because of its half life (29.1 years). The total volume of coolant would require storage for around 300 years to allow decay of strontium to background levels; thus, the tank volume would be in feasibly large.

Abatement Technique Optioneering (Airborne Strontium)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2 –with 2 good, -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For Airborne Strontium** (Green indicates techniques employed in AP1000 NPP)					
	🟩 **Carbon Delay Beds**	**Direct Discharge**	**Wet Scrubbing**	🟩 **HEPA Filtration**		
Proven Technology	2	0	-1	2		
Available Technology	2	0	0	2		
Effective Technology	-2	-2	-2	2		
Ease of Use	2	2	-2	2		
Cost	-1	2	-2	-1		
Impact (Public Dose)	1	-2	0	2		
Impact (Operator Dose)	-1	1	0	-1		
Impact (Environmental)	1	-2	0	2		
Generates Suitable Waste Form	-1	0	0	-1		
Secondary & Decommissioning Waste	-1	0	-1	-1		
Totals	2	-1	-8	8		

Notes on (Gaseous) Abatement Techniques

a) The use of carbon delay beds for strontium is considered impractical because of its half life (29.1 years). The total volume of off-gas would require storage for around 300 years to allow decay of strontium to background levels; thus, the tank volume would be in feasibly large.

b) HEPA filtration is used on the radioactively contaminated area ventilation systems.

RP Predicted Normal Emission (MBq/y)

(including allowance for normal operational fluctuation)

The expected strontium-90 release from an AP1000 NPP is:

0.44 MBq/y to the atmosphere [Reference 4, Table 3.3-8]

0.25 MBq/y as liquid effluent [Reference 4, Table 3.4-6]

4. BAT Assessment Form – Sr-90 (cont.)

RP Predicted Maximum Emission (TBq/y) (including allowance for normal operational fluctuation)	Worst Case Plant Discharge to atmosphere 7.33E-07 TBq/y [Reference 4, Table 6.1-5]. Worst Case Plant Discharge to liquid 5.35E-07 TBq/y [Reference 4, Table 6.1-6]
Comparison with Emissions from Other Nuclear Power Stations	**South Texas 1** 0.186 MBq/y liquid [Reference 6, 5.1.3, p. 33, Table 5-13] **Braidwood 1** 0.414 Bq/y gaseous ND Bq/y liquid [Reference 6, 5.1.3, p. 34, Table 5-14] **Vogtle 1** 138.01 Bq/y gaseous 0.111 MBq/y liquid [Reference 6, 5.1.3, p. 36, Table 5-16]
Associated Sr-90 Dose (µSv/y)	Marine discharge – fisherman family dose 1.5E-06 µSv/y [Reference 4, Table 5.2.12] Aerial discharge – local resident family dose 4.5E-05 µSv/y [Reference 4, Table 5.2.16]
Sr-90 Dose as a Percentage of Total Dose	Marine discharge – fisherman family dose ~0.00007% Aerial discharge – local resident family dose ~0.001%

References

1. “Radio Nuclides Handbook,” R&D Technical Report P3-101/SP1b, Environment Agency, October 2003.
2. EPS-GW-GL-700, Rev 1, “AP1000 European Design Control Document,” Westinghouse Electric Company LLC, 2009.
3. “Radiochemistry in Nuclear Power Reactors,” Commission on Physical Sciences, Mathematics and Applications, 1996.
4. UKP-GW-GL-790, Rev 4, “UK AP1000 Environment Report,” Westinghouse Electric Company LLC, 2011.
5. “Application of Ion Exchange Processes for the Treatment of Radioactive Waste and Management of Spent Ion Exchangers,” Technical Report Series No. 408, International Atomic Energy Agency, Vienna, 2002.
6. APP-WLS-M3C-049, Revision 2 “Monthly Radiation Emissions from Radioactive Nuclides,” AP1000 Calculation Note (proprietary / protect commercial) Westinghouse Electric Company LLC, 2009.

5. BAT Assessment Form – I-131			
Individual Radionuclide Information			
Radionuclide	I-131	**Discharge Route (Solid/Liquid/Gas)**	Gas / Liquid
Radioactive Half-life	8.02 days	**Parent** **Daughter**	N/A Xe - 131
Principal Decay Mode	Beta (gamma)	**Average Beta Energy** **Total Gamma Energy**	0.190 MeV 0.38 MeV
Speciation		Iodine is a halogen element that exhibits a number of stable oxidation states. Two of the most important of these are the -1 (iodide) and +5 (iodate) compounds. Iodine can also take part in the formation of organic complexes. [Reference 1, p. 88] According to the WEC Design Control Document, the iodine form is consistent with the NUREG-1465 model. The model shows the iodine to be predominantly in the form of non-volatile caesium iodide with a small fraction existing as elemental iodine. Additionally, the model assumes that a portion of the elemental iodine reacts with organic materials in the containment to form organic iodine compounds. The resulting iodine species split is as follows: • Particulate 0.95 • Elemental 0.0485 • Organic 0.0015 [Reference 3, subsection 15.6.5.3.1.3]	
Source of Radioactivity (back to the point of generation)		In a PWR reactor Iodine-131 is formed in the fuel by fission and can escape into the reactor coolant water via defects. Escape through defects is accentuated by changes in reactor conditions, particularly reactor power and pressure that occur during operations such as a reactor shutdown. This phenomenon is known as fission product spiking. [Reference 2, p. 24] Even though the reactor core may contain no defective fuel, natural uranium contamination of core construction materials and Zircaloy cladding, as well as enriched uranium contamination of the external cladding surfaces, could be the source of fission products in the coolant during power operations. [Reference 8, subsection 3.1.3]	

<table>
<tr><th colspan="2">5. BAT Assessment Form – I-131 (cont.)</th></tr>
<tr><td>Source Activity (Bq)
(before abatement)</td><td>Design Basis
Reactor Coolant Activity 7.1E-01 μCi/g (26270 Bq/g)
Steam Generator Secondary Side
Liquid Activity 1.1E-03 μCi/g (40.7 Bq/g)
[Reference 3. Table 11.1-2 and 11.1.5]
Realistic Source Terms:
Reactor Coolant Activity 0.04 μCi/g (1480 Bq/g)
Steam Generator Liquid Activity 2.7E-06 μCi/g (0.1 Bq/g)
Steam Generator Steam Activity 2.7E-08 μCi/g (1E-03 Bq/g)
[Reference 3. Table 11.1-8 (sheet 1 of 4)]</td></tr>
<tr><td>Techniques to Prevent or Minimise at Source
(options considered and selected)</td><td>The production of isotopes of iodine, including iodine-131 is an unavoidable consequence of the nuclear fission process.
The reactor is designed and operated so as to minimise the likelihood of damage to the fuel. Leaking fuel pins are located during refueling and are not reused.</td></tr>
<tr><td>Techniques for minimisation at source – Optioneering
(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)</td><td>
<table>
<tr><th rowspan="2">Criteria
(Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)</th><th colspan="4">Technique for minimising production of I-131
(Green indicates techniques employed in AP1000 NPP)</th></tr>
<tr><th>Minimisation of fuel defects in operation</th><th>Control of U contamination in manufacture and fabrication</th><th></th><th></th></tr>
<tr><td>Proven Technology</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Available Technology</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Effective Technology</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Ease of Use</td><td>1</td><td>0</td><td></td><td></td></tr>
<tr><td>Cost</td><td>2</td><td>-1</td><td></td><td></td></tr>
<tr><td>Impact (Public Dose)</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Impact (Operator Dose)</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Impact (Environmental)</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Generates Suitable Waste Form</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Secondary & Decommissioning Waste</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Totals</td><td>19</td><td>15</td><td></td><td></td></tr>
</table>
</td></tr>
</table>

5. BAT Assessment Form – I-131 (cont.)	
Notes on scoring of minimisation techniques	
Pathway to Environment from Source	Iodine-131 enters the environment either as a gaseous emission via the plant ventilation discharge stack, or entrained in liquid effluent. The principal source of iodine-131 is as a fission gas produced in the fuel. Normally, this would be trapped in plena within the fuel pins. However, a migration path to the coolant is created in cases where fuel pins develop defects. The release of iodine to coolant through this route peaks during periods when reactor power is decreased or the reactor is shut down.
Downstream Abatement Techniques (options considered and selected)	**Liquid Abatement Techniques** Mixed Bed Demineralisers In the AP1000 NPP mixed bed demineralisers are provided in the purification loop to remove ionic corrosion products and certain ionic fission products; they also remove zinc during periods of zinc addition.* The demineralisers also act as filters. The mixed bed demineraliser in service can be supplemented by intermittent use of the cation bed demineraliser for additional purification in the event of fuel defects. In this case, the cation resin removes mostly lithium and caesium isotopes. The cation bed demineraliser has sufficient capacity to maintain the cesium-136 concentration in the reactor coolant below 1.0 microcurie per cubic centimetre with design basis fuel defects. Each mixed bed and the cation bed demineraliser is sized to accept the maximum purification flow. Filters are provided downstream of the demineralisers to collect particulates and resin fines. [Reference 3, subsection 9.3.6.2.1.1] * A mixture of lithiated cation and anion resin is used in the demineraliser. Both forms of resin remove fission and corrosion products [Reference 3, subsection 9.3.6.3.4] Chemical Trapping Iodine may be trapped by adding appropriate chemicals (for example, hydrazine hydrate) in the spray system, or by adding chemicals in the reactor sump [Reference 10] **Gaseous/Airborne Iodine Abatement Techniques** Deposition The AP1000 NPP does not include active systems for the removal of activity from the containment atmosphere. The containment atmosphere is depleted of elemental iodine and of particulates as a result of natural processes within the containment. Elemental iodine is removed by deposition onto surfaces. Particulates are removed by sedimentation, diffusiophoresis (deposition driven by steam condensation), and thermophoresis (deposition driven by heat transfer). No removal of organic iodine is assumed. [Reference 3, subsection 15.6.5.3.2]

5. BAT Assessment Form – I-131 (cont.)	
Downstream Abatement Techniques (cont.) (options considered and selected)	Impregnated Charcoal Filters Radioactive iodine arising from power plant operations is routinely removed by impregnated charcoal filters used in combination with particulate filters. Impregnation is required to trap the organic iodine compounds from gas effluents [Reference 9] Delay Beds Carbon delay beds are utilised in the AP1000 NPP. Silver Reactor Experience exists of the Hanford PUREX (Plutonium-Uranium Extraction) silver reactor which was a solid absorber employed primarily to retain 131-I long enough to permit its decay. It utilized beds of Berl saddles (or other similar packing) over which concentrated silver nitrate was poured and evaporated. It was operated at a temperature of around 190°C. Typically, a new bed achieved a RF of 1000, but over time an average of 100 was more demonstrated to be more realistic. Mercurex Process The Mercurex process is a liquid scrubber technology that uses mercuric nitrate-nitric acid scrubbing to complex the iodine and hold it in solution. If the nitric acid is greater than 8 M, then methyl iodide is removed in addition to elemental iodine, which is trapped effectively at lower nitric acid concentrations. The proposed treatment of the waste involves precipitation of mercuric iodate as a first step. Iodox Iodox is a liquid scrubbing method. It employs hyperazeotropic nitric acid (20 to 22 M), which oxidizes all iodine species to the iodate or to HI3O8, which can be recovered by evaporation of the nitric acid. The RFs obtained are high, 1000 to 10000. Electrolytic Scrubbing A liquid scrubbing method, electrolytic scrubbing, employs chemical oxidation to accomplish the same result as the Iodox process; only this method uses an electrolytically generated oxidant such as Co(III). Organic Liquids Organic liquids have been proposed as scrubs because of iodine's high solubility in organic liquids. Organics ranging from fluorocarbons at low temperatures to tributyl phosphate at ambient temperatures have been suggested. Organic Solids Organic solids such as macroreticular resins have also been proposed as iodine sorbers. Most organic polymers have a high affinity for iodine.

5. BAT Assessment Form – I-131 (cont.)

Downstream Abatement Techniques (cont.)

(options considered and selected)

<u>Caustic Scrubbing (utilising sodium or potassium hydroxide)</u>

Caustic scrubbing of iodine from gases is an inexpensive and well-established method for trapping gaseous iodine.

<u>Silver containing sorbents</u>

Three silver-containing solids have been successfully tested for iodine trapping: silver mordenite (AgZ); silver faujasite (AgX); and amorphous silica, which contains silver nitrate (AgSi). The silver zeolites are made by treating the respective zeolites with a silver nitrate solution to replace sodium sites in the framework with silver ions. The AgSi is prepared using a proprietary process.

<u>Note 1</u> – Details of these methods can be found in Reference 11.

<u>Note 2</u> – According to the OSPAR Commission [Reference 4, p. 20] in their national reports, Contracting Parties generally acknowledged that operational management systems are in place to prevent, eliminate or reduce liquid waste. Such systems are an essential element of the application of BAT. In addition, the abatement techniques, identified in the NEA and IAEA reports, on available liquid effluent options, have been employed by Contracting Parties individually or in combination to remove particular materials and nuclides (except tritium and carbon 14) from the liquid effluents. There is a significant level of agreement in the processes being employed, which provides a strong indication that international best practice – and by extension BAT – is being applied.

Abatement Technique Optioneering (I-131 in Liquid)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria **(Scoring -2 to 2 –with 2 good, -2 poor, 0 indicates neither a benefit nor a disadvantage)**	**Abatement Technique For I-131 in Liquid** **(Green indicates techniques employed in AP1000 NPP)**				
	🟩 **Mixed Bed Demineralisers**	**Chemical trapping**			
Proven Technology	**2**	**1**			
Available Technology	**2**	**1**			
Effective Technology	**2**	**1**			
Ease of Use	**2**	**-1**			
Cost	**0**	**-1**			
Impact (Public Dose)	**1**	**1**			
Impact (Operator Dose)	**-1**	**-1**			
Impact (Environmental)	**1**	**1**			
Generates Suitable Waste Form	**-1**	**-2**			
Secondary & Decommissioning Waste	**-1**	**-1**			
Totals	**7**	**-1**			

5. BAT Assessment Form – I-131 (cont.)

Notes on (Liquid) Abatement Techniques

Abatement Technique Optioneering (Airborne I-131)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2 –with 2 good, -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For I-131 in Gas** (Green indicates techniques employed in AP1000 NPP)					
	Deposition	**Delay beds**	**Silver reactor**	**Mercurex Process**	**Iodox**	**Electrolyc Scrubbing**
Proven Technology	2	2	-1	1	2	0
Available Technology	2	2	-1	-2	-2	-2
Effective Technology	2	1	1	1	2	2
Ease of Use	2	2	-2	-2	-2	-2
Cost	0	0	-2	-2	-2	-2
Impact (Public Dose)	2	1	1	1	2	2
Impact (Operator Dose)	2	-1	-2	-2	-2	-2
Impact (Environmental)	2	1	1	1	2	2
Generates Suitable Waste Form	2	0	0	-2	-2	-2
Secondary & Decommissioning Waste	0	0	-2	-2	-2	-2
Total	16	8	-7	-8	-4	-6

Criteria (Scoring -2 to 2 –with 2 good, -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For I-131 in Gas** (Green indicates techniques employed in AP1000 NPP)				
	Organic Liquids	**Organic Solids**	**Caustic Scrubbing**	**Silver containing sorbents**	**Direct discharge**
Proven Technology	-1	-1	2	2	2
Available Technology	-2	-2	2	2	2
Effective Technology	-1	-1	-2	2	-2
Ease of Use	-2	-2	-2	-2	2
Cost	-2	-2	-2	-2	2
Impact (Public Dose)	0	0	0	1	-2
Impact (Operator Dose)	-2	-2	-1	-2	2
Impact (Environmental)	0	0	0	1	-2
Generates Suitable Waste Form	-2	-2	-1	-2	-2
Secondary & Decommissioning Waste	-2	-2	-2	-2	2
Totals	-14	-14	-6	-2	4

5. BAT Assessment Form – I-131 (cont.)	
Notes on (Gaseous) Abatement Techniques	Deposition The AP1000 NPP does not include a safety-related containment spray system to remove airborne particulates or elemental iodine. Removal of airborne activity is by natural processes that do not depend on sprays (that is, sedimentation, diffusiophoresis, and thermophoresis). Much of the non-gaseous airborne activity would eventually be deposited in the containment sump solution. Long-term retention of iodine in the containment sump following design basis accidents requires adjustment of the sump solution pH to 7.0 or above. [Reference 3, Section 6.5.2] Control of the pH in the containment sump water post-accident is achieved through the use of pH adjustment baskets containing granulated trisodium phosphate (TSP). The TSP is designed to maintain the pH of the containment sump water in a range from 7.0 to 9.5. This chemistry reduces radiolytic formation of elemental iodine in the containment sump, consequently reducing the aqueous production of organic iodine, and ultimately reducing the airborne iodine in containment and offsite doses. [Reference 3, Section 6.3.2.1.4]. Silver Reactor The experience of this system at Hanford is that in principle the device is simple and, aside from the cost of silver, inexpensive. In practice, operating problems existed, including difficulties with temperature control. Although the silver reactor could be regenerated with fresh silver nitrate solution, reactor efficiency tended to degenerate with each successive regeneration. The silver reactor has not been tested for 129-I control or long-term iodine retention. Mercurex Process No suitable method for handling the mixed radioactive and hazardous waste has ever been developed. Questions remain about the relative hazards of mercury and iodine, and these potential hazards need to be evaluated before the mercurex process is considered further for use. Iodox The difficulty of working with the highly corrosive liquid has discouraged application. No large-scale testing has been done. Electrolytic Scrubbing Very little development work has been done with this method. Organic Liquids The complexity of the systems, including recovery of the organic and disposal methods for iodine has made the methods unattractive. Organic liquids have not been employed on a plant scale.

<table>
<tr><th colspan="2">5. BAT Assessment Form – I-131 (cont.)</th></tr>
<tr><td>Notes on (Gaseous) Abatement Techniques (cont.)</td><td>Organic Solids

The iodine is easily desorbed. In addition, waste treatment of radioactive organic materials is difficult. Organic solids have not been employed on a plant scale.

Caustic Scrubbing (utilising sodium or potassium hydroxide)

The reduction by scrubbing may be small as it is predicted to be an inefficient process at the low iodine concentrations involved for very fine particulates and for organic iodides.

Silver containing sorbents (for example, silver substituted zeolites, silver-nitrate impregnated amorphous silica)

Disposal of iodine-loaded silver zeolites has been studied at Pacific Northwest National Laboratory. Incorporation in concrete was identified as the “best” available disposal technique but there is no practical containment method that will prevent iodine release after a period of a few hundred years. The disposal of the iodine-loaded silver zeolites is complicated by classification of silver as a hazardous material requiring treatment of this waste form as a mixed hazardous radioactive waste.</td></tr>
<tr><td>RP Predicted Normal Emission (GBq/y)

(including allowance for normal operational fluctuation)</td><td>The expected I-131 release from an AP1000 NPP is 0.21GBq/y to the atmosphere [Reference 12, Table 2-6]

The expected I-131 release from an AP1000 NPP is 0.015 GBq/y to coastal water [Reference 12, Table 2-2]</td></tr>
<tr><td>RP Predicted Maximum Emission (TBq/y)

(including allowance for normal operational fluctuation)</td><td>Worst Case Plant Discharge to atmosphere 3.42E-04 TBq/y [Reference 5, Table 6.1-5]</td></tr>
<tr><td>Comparison with Emissions from Other Nuclear Power Stations</td><td>Sizewell B – Predicted

The predicted future rolling 12-month discharge of Iodine – 131 is 1 GBq to the atmosphere [Reference 2, p. 87]

Mean and standard deviation of the data available for predecessor designs (Airborne iodine-131) [Reference 6, Table 2]
<table>
<tr><th>Design</th><th>Mean GBq/Gweh</th><th>Standard GBq/Gweh deviation</th><th>Maximum GBq/Gweh</th><th>Predicted GBq/Gweh</th></tr>
<tr><td>AP1000 NPP</td><td>1.35E-05</td><td>4.34E-05</td><td>5.69E-05</td><td>4.54E-04</td></tr>
<tr><td>EPR</td><td>1.05E-06</td><td>1.95E-06</td><td>3.00E-06</td><td>1.57E-06</td></tr>
<tr><td>ESBWR</td><td>2.82E-06</td><td>5.24E-06</td><td>8.06E-06</td><td>1.10E-03</td></tr>
<tr><td>ACR1000</td><td>5.66E-06</td><td>1.10E-05</td><td>1.67E-05</td><td>8.42E-07</td></tr>
</table></td></tr>
</table>

5. BAT Assessment Form – I-131 (cont.)

Normalised releases of radionuclides (iodine-131) from nuclear reactors (TBq/Gwy) [Reference 7, Table 37]

Year	PWR	BWR	HWR
1970-1974	0.0033	0.15	0.0014
1975-1979	0.0050	0.41	0.0031
1980-1984	0.0018	0.093	0.0002
1985-1989	0.0009	0.0018	0.0002
1990-1994	0.0003	0.0008	0.0004
1995-1997	0.0002	0.0003	0.0001

Impact (µSv/y) (in terms of dose to human and non-human species)	Aerial discharge – local resident family dose 1.3E-01 µSv/y [Reference 5, Table 5.2.16]
I-131 Dose as a Percentage of Total Dose	Aerial discharge – local resident family dose ~3%

References

1. “Radionuclides Handbook,” R&D Technical Report P3-101/SP1b, Environment Agency, October 2003.
2. “Review of the Control and Impact of the Discharge and Disposal of Radioactive Waste at Sizewell B Power Station,” SZB/THR/042, Revision 000, British Energy Generation Ltd.
3. EPS-GW-GL-700, Rev 1, “AP1000 European Design Control Document,” Westinghouse Electric Company LLC, 2009.
4. “Assessment of the 4th Round of Reporting on the Implementation of PARCOM Recommendations 91/4 on Radioactive Discharges,” OSPAR Commission, ISBN 978-1-905859-90-0.
5. UKP-GW-GL-790, Rev 4, “UK AP1000 Environment Report,” Westinghouse Electric Company LLC, 2011.
6. “Study of historic nuclear reactor discharge data,” Coppleston, D. et al., Radioprotection, Vol 44, No 5, (2009) 875 – 880.
7. UNSCEAR 2000 Report, Vol 1, Annex C, United Nations Scientific Committee on the Effects of Atomic Radiation.
8. “Radiochemistry in Nuclear Power Reactors,” Commission on Physical Sciences, Mathematics and Applications, 1996.
9. “Radioactive waste management at nuclear power plants,” V M Efremenkov, IAEA Bulletin 4/1989.
10. “Radiation Protection Aspects of Design for Nuclear Power Plants,” Safety Guide No NS-G-1.13, IAEA Safety Standards.
11. HWVP Iodine Trap Evaluation, Burger, L. L., Scheele, R. D., PNNL-14860, Pacific Northwest National Laboratory, September 2004.
12. APP-WLS-M3C-040, Revision 0, “Expected Radioactive Effluents Associated with Advanced Plant Designs,” Westinghouse Electric Company LLC, 2008.

<table>
<tr><th colspan="4">6. BAT Assessment Form – Cs-137</th></tr>
<tr><td colspan="4">Individual Radionuclide Information</td></tr>
<tr><td>Radionuclide</td><td>Cs-137</td><td>Discharge Route (Solid/Liquid/Gas)</td><td>Gas / Liquid</td></tr>
<tr><td>Radioactive Half-life</td><td>30 years</td><td>Parent
Daughter</td><td>N/A
Ba – 137 m [R]</td></tr>
<tr><td>Principal Decay Mode</td><td>Beta (gamma)</td><td>Average Beta Energy
Total Gamma Energy</td><td>0.187 MeV</td></tr>
<tr><td colspan="2">Speciation</td><td colspan="2">Caesium is an alkali metal whose chemical behaviour is determined by the properties of the Cs$^+$ ion.
Most of the compounds of caesium are ionic in nature, although more complex species can be formed.
Caesium reacts extremely vigorously with water, oxygen and halogens.
[Reference 1, p.88]</td></tr>
<tr><td colspan="2">Source of Radioactivity
(back to the point of generation)</td><td colspan="2">Caesium-137 is a fission product and emits both beta and gamma radiation. A major fuel failure would result in large amounts of Caesium in the reactor cooling water, and then it would be a more significant component of the liquid effluent [Reference 2, p. 23].
In theory, the measurements of Cs-134 and Cs-137 activities in the reactor coolant may be used as an indicator of the burnup of a failed fuel rod from which the fission products are released. Cs-137 is produced directly from fission, and Cs-134 is produced by neutron activation of Cs-133, a stable fission product. Since the Cs-134 activity increases (proportionally to the square of the fuel burnup) faster than Cs-137 in the fuel as the fuel burnup increases, the ratio of Cs-134 to Cs-137 increases as the fuel burnup increases.
Even though the reactor core may contain no defective fuel, natural uranium contamination of core construction materials and Zircaloy cladding, as well as enriched uranium contamination of the external cladding surfaces, could be the source of fission products in the coolant during power operations. [Reference 5, Section 3.1.3]</td></tr>
<tr><td colspan="2">Source Activity (Bq)
(before abatement)</td><td colspan="2">Design Basis
Reactor Coolant Activity 5.0E-01 µCi/g (18500 Bq/g)
Steam Generator Secondary Side
Liquid Activity 1.5E-03 µCi/g (55.5 Bq/g)
[Reference 3, Table 11.1-2 and 11.1.5]</td></tr>
</table>

<table>
<tr><th colspan="2">6. BAT Assessment Form – Cs-137 (cont.)</th></tr>
<tr><td></td><td>Realistic Source Terms:
Reactor Coolant Activity 7.9E-03 µCi/g (292.3 Bq/g)
Steam Generator Liquid Activity 2.0E-06 µCi/g (0.074 Bq/g)
Steam Generator Steam Activity 9.9E-09 µCi/g (3.66E-04 Bq/g)
[Reference 3, Table 11.1-8 (sheet 2 of 4)]</td></tr>
<tr><td>Techniques to Prevent or Minimise at Source
(options considered and selected)</td><td>The production of isotopes of caesium, including caesium-137, is an unavoidable consequence of the nuclear fission process.
Leaking fuel pins are the prime route for caesium isotope transmission to coolant. The reactor is designed and operated so as to minimise the likelihood of damage to the fuel. Leaking fuel pins are located during refuelling and are not reused.</td></tr>
<tr><td>Techniques for minimisation at source – Optioneering
(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)</td><td>
<table>
<tr><th rowspan="2">Criteria
(Scoring -2 to 2, with 2 good and-2 poor, 0 indicates neither a benefit nor a disadvantage)</th><th colspan="4">Technique for minimisation production of Cs-137
(Green indicates techniques employed in AP1000 NPP)</th></tr>
<tr><th>Minimisation of fuel defects in operation</th><th>Control of U contamination in manufacture and fabrication</th><th></th><th></th></tr>
<tr><td>Proven Technology</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Available Technology</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Effective Technology</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Ease of Use</td><td>1</td><td>0</td><td></td><td></td></tr>
<tr><td>Cost</td><td>2</td><td>-1</td><td></td><td></td></tr>
<tr><td>Impact (Public Dose)</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Impact (Operator Dose)</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Impact (Environmental)</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Generates Suitable Waste Form</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Secondary & Decommissioning Waste</td><td>2</td><td>2</td><td></td><td></td></tr>
<tr><td>Totals</td><td>19</td><td>15</td><td></td><td></td></tr>
</table>
</td></tr>
<tr><td>Notes on scoring of minimisation techniques</td><td></td></tr>
<tr><td>Pathway to Environment from Source</td><td>Caesium-137 enters the environment entrained in liquid effluent.
The principal source of caesium-137 is as a fission product produced in the fuel. Normally, this would be trapped within the fuel pins. However, a migration path to the coolant is created in cases where fuel pins develop defects.</td></tr>
</table>

6. BAT Assessment Form – Cs-137 (cont.)	
Pathway to Environment from Source (cont.)	Caesium-137 can accumulate in living organisms and sediments. The principal pathway for public radiation dose is through the ingestion of local fish and shellfish, and possibly by occupying inter-tidal areas where there may be sediment deposits.
Downstream Abatement Techniques (options considered and selected)	**Liquid Abatement Techniques** Demineralisers In the AP1000 NPP zeolite, cation exchange and mixed bed demineralisers are provided in the purification loop to remove ionic corrosion products and certain ionic fission products; they also remove zinc during periods of zinc addition.* The demineralisers also act as filters. The mixed bed demineraliser in service can be supplemented by intermittent use of the cation bed demineraliser for additional purification in the event of fuel defects. In the case of caesium the zeolite bed and cation resin are most effective at removing the caesium isotopes. The cation bed demineraliser has sufficient capacity to maintain the cesium-136 concentration in the reactor coolant below 1.0 microcurie per cubic centimetre with design basis fuel defects. Each mixed bed and the cation bed demineraliser is sized to accept the maximum purification flow. Filters are provided downstream of the demineralisers to collect particulates and resin fines. [Reference 3, subsection 9.3.6.2.1.1] *A mixture of lithiated cation and anion resin is used in the demineraliser. Both forms of resin remove fission and corrosion products [Reference 3, subsection 9.3.6.3.4] During normal operation, the reactor coolant contains lithium hydroxide. The demineraliser in the CVS used to routinely clean-up reactor coolant on-load is saturated with lithium ions, making it less effective at removing some radionuclides including caesium-137. However, with the reactor shutdown for refuelling, there are no longer significant amounts of lithium hydroxide left in the reactor coolant, and it is possible to use an alternative demineraliser bed that is not saturated with lithium ions to perform more effective clean-up of the reactor coolant and Caesium-137. When letdown is being diverted to the liquid radwaste system, the purification flow is routed through the cation bed demineraliser for removal of as much lithium-7 and cesium as possible. [Reference 3, subsection 9.3.6.2.3.2] One cation resin bed demineraliser is located downstream of the mixed bed demineralisers and is used intermittently to control the concentration of lithium-7 (pH control) in the RCS. The demineraliser is sized to accommodate maximum purification flow when in service, which is adequate to control the lithium-7 and/or cesium concentration in the reactor coolant. [Reference 3, subsection 9.3.6.3.4] Filtration Filtration is not consider a viable option for separation of water soluble Cs.

6. BAT Assessment Form – Cs-137 (cont.)

Downstream Abatement Techniques (cont.) (options considered and selected)	**Direct Discharge**

Abatement Technique Optioneering (Cs-137 in Liquid)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique for Cs-137 in Liquid** (Green indicates techniques employed in AP1000 NPP)								
	Demineralises	**Filtration**	**Direct Discharge**						
Proven Technology	2	-2	2						
Available Technology	2	-2	2						
Effective Technology	2	-2	-2						
Ease of Use	2	-2	2						
Cost	0	0	2						
Impact (Public Dose)	2	-2	-2						
Impact (Operator Dose)	-1	2	2						
Impact (Environmental)	2	-2	-2						
Generates Suitable Waste Form	2	-2	-2						
Secondary & Decommissioning Waste	-1	2	2						
Totals	12	-10	4						

Notes on (Liquid) Abatement Techniques

Demineralisers – Assumed Decontamination factors for caesium:

Zeolite bed	100
Cation resin	10
Mixed resin 1	2
Mixed resin 2	10
Overall removal efficiency	>99%

6. BAT Assessment Form – Cs-137 (cont.)	
RP Predicted Normal Emission (GBq/y) (including allowance for normal operational fluctuation)	The expected caesium – 137 release from an AP1000 NPP is 0.0013GBq/y to the atmosphere [Reference 6, Table 2-8]. The expected caesium – 137 release from an AP1000 NPP is 0.023 GBq/y to coastal water [Reference 6, Table 2-2]
RP Predicted Maximum Emission (GBq/y) (including allowance for normal operational fluctuation)	Worst Case Plant Discharge to atmosphere 2.20E-06 TBq/y [Reference 4, Table 6.1-5] Worst Case Plant Discharge to liquid 4.97E-05 TBq/y [Reference 4, Table 6.1-6]
Comparison with Emissions from Other Nuclear Power Stations	**Sizewell B – Predicted** The predicted future rolling 12-month discharge of caesium – 137 is 5 - 13 GBq. [Reference 2, p. 100]
Impact (μSv/y) (in terms of dose to human and non-human species)	Marine discharge - fisherman family dose 3.4E-03 μSv/y [Reference 4, Table 5.2.12] Aerial discharge – local resident family dose 1.3E-04 μSv/y [Reference 4, Table 5.2.16]
Cs-137 Dose as a Percentage of Total Dose	Marine discharge – fisherman family dose ~0.1% Aerial discharge – local resident family dose ~0.005%

References

1. “Radionuclides Handbook,” R&D Technical Report P3-101/SP1b, Environment Agency, October 2003.
2. “Review of the Control and Impact of the Discharge and Disposal of Radioactive Waste at Sizewell B Power Station,” SZB/THR/042, Revision 000, British Energy Generation Ltd.
3. EPS-GW-GL-700, Rev 1, “AP1000 European Design Control Document,” Westinghouse Electric Company LLC, 2009.
4. UKP-GW-GL-790, Rev 4, “UK AP1000 Environment Report,” Westinghouse Electric Company LLC, 2011.
5. Radiochemistry in Nuclear Power Reactors. Commission on Physical Sciences, Mathematics and Applications, 1996.
6. APP-WLS-M3C-040 Revision 0, “Expected Radioactive Effluents Associated with Advanced Plant Designs,” Westinghouse Electric Company LLC, 2008.

7. BAT Assessment Form Pu-241

Individual Radionuclide Information			
Radionuclide	Pu-241	**Discharge Route (Solid/Liquid/Gas)**	Liquid
Radioactive Half-life	14.4 years	**Parent** **Daughter**	Cm-245 Am-241[R]
Principal Decay Mode	Beta	**Average Energy**	0.005 MeV
Speciation		In aqueous solution, plutonium can exhibit any of four oxidation states. The stable oxidation state(s) in any solution are a function of environmental conditions such as pH and Eh. Plutonium reacts slowly with water and rapidly with dilute acids. It forms halide and oxide compounds. [Reference 1, p. 144]	
Source of Radioactivity (back to the point of generation)		The transuranic elements are those produced by successive neutron capture of uranium and its products in a reactor. Included in the major products of the production chain is the Pu-241 isotope. [Reference 6, p. 18] The production of Pu-241 is an inevitable consequence of uranium fission reactions. Radionuclides may enter the RCS is as a result of fuel leaks. This leakage is very low. [Reference 2, subsection 2.1.11]	
Source Activity (Bq) (before abatement)		**Design Basis** No data listed [Reference 3]	
Techniques to Prevent or Minimise at Source (options considered and selected)		Pu-241 production is unavoidable in systems using uranium as a fuel. The ability to completely retain Pu-241 within fuel pins cannot be guaranteed, although improvements in clad material and quality control during manufacture have greatly reduced the incidence of pin failures. Good operation also contributes to this.	

7. BAT Assessment Form – Pu-241 (cont.)

Techniques for minimisation at source – Optioneering

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2, with 2 good and-2 poor, 0 indicates neither a benefit nor a disadvantage)	**Technique for minimisation production of Pu-241 isotope** (Green indicates techniques employed in AP1000 NPP)				
	🟩 **Ultrasonic fuel cleaning**	🟩 **Minimise plant shutdowns**	🟩 **Material Selection and QA/QC**		
Proven Technology	2	2	2		
Available Technology	2	2	2		
Effective Technology	2	2	2		
Ease of Use	-1	1	2		
Cost	-1	1	-2		
Impact (Public Dose)	1	1	2		
Impact (Operator Dose)	1	1	2		
Impact (Environmental)	1	1	2		
Generates Suitable Waste Form	1	1	2		
Secondary & Decommissioning Waste	2	1	2		
Totals	10	13	16		

Notes on scoring of minimisation techniques

None

Pathway to Environment from Source

The main anticipated release route for Pu-241 is by liquid discharge following treatment.

Downstream Abatement Techniques

(options considered and selected)

a) The liquid radwaste system comprising filtration and ion exchange beds is effective in reducing plutonium levels in liquid effluents.

b) Gaseous Filtration on discharge outlets to atmosphere minimises emissions. [Reference 2, subsection 2.2.1]

c) The Fuel Storage Pond Cooling and Clean-up System (FSPCCS) is designed to control contamination of the fuel storage pond and ensure that heat from the fuel is removed. The water is re-circulated to maintain the required chemical and radioactivity conditions, so the radioactivity transferred to the LRWS for discharge to the environment

7. BAT Assessment Form – Pu-241 (cont.)

Downstream Abatement Techniques (cont.)

(options considered and selected)

is minimised. For the radioactivity that originates from the ponds, minimisation of waste relates initially to the minimisation of contamination of the pond water, and its discharge. The fuel storage pond water chemistry is controlled to minimise fuel-clad corrosion. All these measures minimise the release of radioactivity into the pond water. [Reference 2, subsection 2.5.4]

d) Monitoring of Discharges. All liquid waste releases are monitored by a radiation monitor prior to discharge. The monitor is located on the common discharge line downstream of the WLS monitor tanks limits for radionuclide concentrations in liquid effluents discharged into unrestricted areas. These radiation monitors will provide a signal to terminate liquid radwaste releases if the discharge concentration in the line exceeds a predetermined set point. (Note – not strictly an abatement process but included for completeness) [Reference 4]

Abatement Technique Optioneering (Pu-241)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

	Abatement Technique For Pu-241 in Liquid (Green indicates techniques employed in AP1000 NPP)								
Criteria (Scoring -2 to 2, with 2 good and-2 poor, 0 indicates neither a benefit nor a disadvantage)	🟩 **Filtration/Ion Exchange**	**Evaporators**	🟩 **FSPCSS**	🟩 **Monitoring of Discharges**	**Delay Tank**	**Adsorption**	**Wet Scrubbing**	**Direct Discharge**	**Precipitation**
Proven Technology	2	2	2	2	2	1	1	0	1
Available Technology	2	2	2	2	2	-1	-1	0	-1
Effective Technology	2	1	2	2	-2	1	1	-2	-1
Ease of Use	-1	-2	-1	0	-2	-2	-2	2	-2
Cost	-1	-2	-1	-1	-2	-2	-2	2	-2
Impact (Public Dose)	2	1	1	1	2	1	1	-2	1
Impact (Operator Dose)	-1	-2	-1	0	-2	-2	-2	2	-2
Impact (Environment)	2	1	1	1	2	1	1	-2	1
Generates Suitable Waste Form	2	1	2	0	2	0	0	-2	-1
Secondary & Decommissioning Waste	-1	-2	-1	0	-2	-2	-2	2	-2
Totals	8	0	6	7	0	-5	-5	0	-8

7. BAT Assessment Form – Pu-241 (cont.)	
Notes on (Liquid) Abatement Techniques	a) The use of delay tanks for Pu-241 is considered impractical because of its half life (14.4 years). The total volume of coolant would require storage for around 144 years to allow decay of Pu-241 to background levels thus the tank volume would be in feasibly large. b) Adsorption – could be applied, but not as effective as ion-exchange c) Wet Scrubbing – applicable to particulate wastes d) Direct discharge not considered acceptable – pre-treatment is carried out in the AP1000 NPP e) Evaporation – could be applied, but implementation costs and (historically) operational difficulties make this an expensive option for removal of very small amounts of Pu-241 f) Precipitation – could be applied, but not considered as effective as ion-exchange
RP Predicted Normal Emission (MBq/y) (including allowance for normal operational fluctuation)	The expected Pu-241 release from an AP1000 NPP is: 0.0814 MBq/y as liquid effluent [Reference 5, 4.5, p. 11 and 5.1.1.12, Table 5-12]
RP Predicted Maximum Emission (MBq/y) (including allowance for normal operational fluctuation)	Worst Case Plant Discharge to liquid 1.78E-07 TBq/y [Reference 7, Table 6.1-6]
Comparison with Emissions from Other Nuclear Power Stations	No comparative isotope data available [Reference 5]
Associated Pu-241 Dose (µSv/y)	Marine discharge – local resident family dose 2.76E-06 µSv/y [Reference 5, Table 5.2.12]
Pu-241 Dose as a Percentage of Total Dose	Marine discharge – local resident family dose ~0.0001%

7. BAT Assessment Form – Pu-241 (cont.)

References

1. "Radionuclides Handbook," R&D Technical Report P3-101/SP1b, Environment Agency, October 2003.
2. "Review of the Control and Impact of the Discharge and Disposal of Radioactive Waste at Sizewell B Power Station," SZB/THR/042, Revision 000, British Energy Generation Ltd.
3. EPS-GW-GL-700, Rev 1, "AP1000 European Design Control Document," Westinghouse Electric Company LLC, 2009.
4. "AP1000 Water Chemistry," K. Newmyer, Westinghouse Electric Company Presentation, 23 July 2008.
5. APP-WLS-M3C-049, Revision 2 "Monthly Radiation Emissions from Radioactive Nuclides," AP1000 Calculation Note (proprietary / protect commercial) Westinghouse Electric Company LLC, 2009.
6. "Radiochemistry in Nuclear Power Reactors," Commission on Physical Sciences, Mathematics and Applications, 1996.
7. UKP-GW-GL-790, Rev 4, "UK AP1000 Environment Report," Westinghouse Electric Company LLC, 2011.

8. BAT Assessment Form Noble Gases			
Individual Radionuclide Information			
Radionuclide	Ar-41 Kr-85 Xe–133	**Discharge Route (Solid/Liquid/Gas)**	Gas; Gas; Gas
Radioactive Half-life	109 mins 10.7 years 5.2 days	**Parent/Daughter**	N/A / K-41 N/A / Rb-85; N/A / Cs-133;
Principal Decay Mode	Beta [Gamma] Beta; Beta;	**Average Beta Energy** **Total Gamma Energy**	0.48; 1.28 MeV 0.25; 0.002 MeV; 0.135; 0.048 MeV
Speciation		Argon, krypton and xenon are noble gases and, as such, form only a limited number of chemical compounds (such as fluorine compounds) due to their lack of reactivity. [Reference 1 – p. 100, 198; 32]	
Source of Radioactivity (back to the point of generation)		Argon-41 is formed as a result of the activation of natural Argon-40 in the air by neutrons close to the reactor vessel. For this reason, its production rate is directly linked to the neutron flux in this region and thus the power level. During reactor operation, krypton-85 and xenon-133 are created as fission products. A portion of these gases are released to the reactor coolant because of a small number of fuel cladding defects. [Reference 3, Section 11.3]	
Source Activity (Bq) (before abatement)		**Design Basis** Reactor Coolant Activity: Ar – 41 ND Kr – 85m 8.4E-01 μCi/g [31080 Bq/g] Kr – 85 3.0 μCi/g [111000 Bq/g] Xe – 133m 1.7 μCi/g [62900 Bq/g] Xe – 133 1.2E+02 μCi/g [4440000 Bq/g] [Reference 3, Table 11.1-2]	

<table>
<tr><th colspan="2">8. BAT Assessment Form Noble Gases (cont.)</th></tr>
<tr><td>Source Activity (Bq) (cont.)
(before abatement)</td><td>Realistic Source Terms
Reactor Coolant Activity:
Ar – 41 ND
Kr – 85m 0.21 µCi/g [7770 Bq/g]
Kr – 85 1.4 µCi/g [51800 Bq/g]
Xe – 133m 1.1 µCi/g [40700 Bq/g]
Xe – 133 0.093 µCi/g [3441 Bq/g]
[Reference 3, Table 11.1-8]</td></tr>
<tr><td>Techniques to Prevent or Minimise at Source
(options considered and selected)</td><td>Leakage of reactor coolant (which contains entrained noble gases) results in a leakage to the containment atmosphere of the noble gases. Airborne releases can be limited both by restricting reactor coolant leakage and by limiting the concentrations of radioactive noble gases in the RCS [Reference 3, Section 11.3].</td></tr>
<tr><td>Techniques for minimisation at source – Optioneering
(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)</td><td>
<table>
<tr><th rowspan="2">Criteria
(Scoring -2 to 2, with 2 good and-2 poor, 0 indicates neither a benefit nor a disadvantage)</th><th colspan="5">Technique for minimising production of tritium isotope
(Green indicates techniques employed in AP1000 NPP)</th></tr>
<tr><th>🟩 Minimisation of fuel defects in operation</th><th></th><th></th><th></th><th></th></tr>
<tr><td>Proven Technology</td><td>2</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Available Technology</td><td>2</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Effective Technology</td><td>2</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Ease of Use</td><td>1</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Cost</td><td>2</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Public Dose)</td><td>2</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Operator Dose)</td><td>2</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Impact (Environmental)</td><td>2</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Generates Suitable Waste Form</td><td>2</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Secondary & Decommissioning Waste</td><td>2</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Totals</td><td>19</td><td></td><td></td><td></td><td></td></tr>
</table>
</td></tr>
<tr><td>Notes on scoring of minimisation techniques</td><td>No comments</td></tr>
</table>

8. BAT Assessment Form Noble Gases (cont.)	
Pathway to Environment from Source	**Gaseous Pathway** The gaseous discharge route is described in Figures 11.3-1 and 11.3-2 of the AP1000 Design Control Document [Reference 3]. Gaseous noble gases are discharged to the atmosphere via the main ventilation stack. The activation of natural Argon-40 to produce Argon-41 occurs within the containment area of the reactor and contributes to airborne contamination. Due to the open structure and the ventilation requirements in the reactor building, Argon-41 may be responsible for external gamma exposure when individuals enter the containment while the reactor is in operation. During normal power operation, excessive activity buildup in the containment atmosphere is prevented by periodic purging of the containment to the plant vent. When the plant is shut down for refuelling or maintenance, additional purging of the containment atmosphere may be performed to further reduce the activity levels consistent with the increased level of worker presence in the containment. Removal of the noble gases from the RCS is not normally necessary because the gases will not build up to unacceptable levels when fuel defects are within normally anticipated ranges. If noble gas removal is required because of high RCS concentration, the CVS can be operated in conjunction with the WLS degasifier to remove the gases. Noble gases pass into the radioactive waste systems by out-gassing from the coolant during normal operation, especially in the Volume Control Tank (VCT) which is connected to the RCS. Reactor coolant that is letdown into the WLS systems will also release noble gases. **Liquid Pathway** Because of the degasification processes described above noble gas release in the AP1000 NPP liquid effluents can be considered to be negligible. [Reference 4, p. 26 and Table 5-9]
Downstream Abatement Techniques (options considered and selected)	The principal measures for reducing the release of noble gases relate to the quality of the fuel cladding and the minimisation of fuel defects. Noble gases are readily stripped out of the liquid effluent by the WLS degasifier and transferred to the gaseous radwaste system. However, neither the liquid nor gaseous radwaste systems are able to remove noble gases from the emission streams since these gases are chemically inert. The carbon delay beds in the gaseous radwaste system reduce shorter lived noble gas emissions (Xe and Ar) but are ineffective for reduction of Kr activity because of its relatively long half-life. Since the noble gases are difficult to separate chemically physical methods need to be considered. One such method is cryogenics which could be used to distil individual gas components. However, the capital and operating costs outweigh the benefits.

8. BAT Assessment Form Noble Gases (cont.)

Abatement Technique Optioneering (Noble Gases)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For Noble Gases** (Green indicates techniques employed in AP1000 NPP)				
	Carbon Delay Beds	**Minimise plant shutdowns**	**Cryogenics**	**Direct Discharge**	
Proven Technology	2	2	2	2	
Available Technology	2	2	2	2	
Effective Technology	2	2	2	2	
Ease of Use	1	1	-2	2	
Cost	-1	0	-2	2	
Impact (Public Dose)	2	1	2	-2	
Impact (Operator Dose)	-1	1	-2	2	
Impact (Environmental)	2	1	2	-2	
Generates Suitable Waste Form	1	0	-2	0	
Secondary & Decommissioning Waste	-1	0	-2	0	
Totals	9	10	0	8	

Notes on (Gaseous) Abatement Techniques

a) Delay beds are effective for the shorter lived noble gases Ar-41 and Xe-133 since the design 60 day hold-back period in these beds is sufficient to significantly reduce the activity levels of these isotopes (Ar-41 will be effectively eliminated).

b) The use of delay beds for Kr-85 is considered impractical because of its relatively long half life (10.7 years). A delay period of around 107 years would be needed to allow decay of tritium to background levels. The tanks would be unfeasibly large to accommodate the amount of krypton bearing gas for the delay period required.

c) Direct discharge currently considered best option for Kr-85 in absence of alternative technical viable or economically practicable alternative

d) Cryogenic systems could be used to liquefy noble gases as part of a separation process. However, it is expensive both in terms of capital and operational costs. The use of complex equipment will result in higher operator dose, and it is likely to result in increased wastes.

e) Plant operation can significantly affect the amount of noble gases released from fuel. Therefore, good plant and fuel design, quality in manufacture, optimising plant availability, good training of operators, and the like are relevant contributors to minimisation of noble gas release.

8. BAT Assessment Form Noble Gases (cont.)	
RP Predicted Normal Emission (GBq/y) (including allowance for normal operational fluctuation)	The expected noble gas releases from an AP1000 NPP are: Ar-41 1.3E+03 Kr-85m 2.4E+01 Kr-85 3.1E+03 Xe-133m 1.1E+02 Xe-133 1.3E+03 [Reference 4, Table 2-7]
RP Predicted Maximum Emission (TBq/y) (including allowance for normal operational fluctuation)	Worst Case Plant Discharge to atmosphere 13.363 TBq/y [Reference 5, Table 6.1-5
Comparison with Emissions from Other Nuclear Power Stations	**South Texas 1** Ar-41 0.222TBq/y gaseous Kr-85 ND gaseous Kr-85m 0.023 TBq/y gaseous Xe-133 5.18 TBq/y gaseous Xe-133 ND gaseous Ar-41 ND liquid Kr-85 ND liquid Kr-85m ND liquid Xe-133 0.173GBq/y liquid Xe-133m 3.64MBq/y liquid [Reference 4, 5.1.3, p.33, Table 5-13] **Braidwood 1** Ar-41 0.0179TBq/y gaseous Kr-85 ND gaseous Kr-85m ND gaseous Xe-133 9.8 GBq/y gaseous Xe-133m 0.042GBq/y gaseous Ar-41 0.126MBq/y liquid Kr-85 0.075 GBq/y liquid Kr-85m ND liquid

8. BAT Assessment Form Noble Gases (cont.)	
Comparison with Emissions from Other Nuclear Power Stations (cont.)	Xe-133 0.310 GBq/y liquid Xe-133m 0.444 MBq/y liquid [Reference 4, 5.1.3, p. 34, Table 5-14] **Cook 1** Ar-41 0.077 TBq/y gaseous Kr-85 6.993 TBq/y gaseous Kr-85m 0.04GBq/y gaseous Xe-133 1.055 TBq/y gaseous Xe-133m 0.045GBq/y gaseous Ar-41 ND liquid Kr-85 ND liquid Kr-85m ND liquid Xe-133 1.798 MBq/y liquid Xe-133m ND liquid [Reference 4, 5.1.3, p. 35, Table 5-15] **Vogtle 1** Ar-41 0.14 TBq/y gaseous - elevated Kr-85 0.108 TBq/y gaseous - elevated Kr-85m 1.42 MBq/y gaseous - elevated Xe-133 0.744 TBq/y gaseous - elevated Xe-133m 0.001 TBq/y gaseous - elevated Ar-41 0.025 GBq/y gaseous - ground Kr-85 ND Kr-85m ND Xe-133 0.751 GBq/y gaseous - ground Xe-133m 3.019 TBq/y gaseous - ground Ar-41 ND liquid Kr-85 ND liquid Kr-85m 0.020 GBq/y liquid Xe-133 0.112 GBq/y liquid Xe-133m ND liquid [Reference 4, 5.1.3, p. 36, Table 5-16]

8. BAT Assessment Form Noble Gases (cont.)

Sizewell B - Predicted

Noble gases	80 TBq/y gaseous
Noble gases	ND liquid

[Reference 4, 5.1.3, p. 37, Table 5-17, and Reference 2]

Comparison of AP1000 NPP Noble Gas Discharges with Average Normalised Releases from European Nuclear Power Plants Between 1995 and 1997

	Unit	AP1000 NPP	Sizewell B	All PWR	All Magnox and AGR	All BWR
Average	TBq/ GWa	7.98	4.36 *	13	463	171
Maximum	TBq/ GWa	26.41	n/a	n/a	n/a	n/a

*Average of 1996, 1997 reported data.

[Reference 6, Table 31]

Associated Noble Gas Dose (µSv/y)

Aerial discharge – local resident family dose

Ar-41	2.9E-02 µSv/y
Kr-85	3.7E-04 µSv/y
Xe-133	6.4E-04 µSv/y

[Reference 5, Table 5.2.16]

Noble Gas Dose as a Percentage of Total Dose

Aerial discharge – local resident family dose

Ar-41 ~1.7% Kr-85 ~ 0.02% Xe-133 ~ 0.04%

References

1. “Radionuclides Handbook,” R&D Technical Report P3-101/SP1b, Environment Agency, October 2003.
2. “Review of the Control and Impact of the Discharge and Disposal of Radioactive Waste at Sizewell B Power Station,” SZB/THR/042, Revision 000, British Energy Generation Ltd.
3. EPS-GW-GL-700, Rev 1, “AP1000 European Design Control Document,” Westinghouse Electric Company LLC 2009.
4. APP-WLS-M3C-049, Revision 2 “Monthly Radiation Emissions from Radioactive Nuclides,” AP1000 Calculation Note (proprietary / protect commercial) Westinghouse Electric Company LLC, 2009.
5. UKP-GW-GL-790, Rev 4, “UK AP1000 Environment Report,” Westinghouse Electric Company LLC, 2011.
6. “UNSCEAR 2000 Report,” Vol 1, Annex C. United Nations Scientific Committee on the Effects of Atomic Radiation.

9. BAT Assessment Form – Beta Particulates

Individual Radionuclide Information			
Radionuclide	Co-58	**Discharge Route (Solid/Liquid/Gas)**	Gas / Liquid
Radioactive Half-life	71 days	**Parent**	N/A
		Daughter	Fe-58
Principal Decay Mode	Beta [gamma]	**Average Energy**	0.034 MeV
		Total Gamma Energy	0.975 MeV
Speciation		Cobalt is a transition metal element that shows two common oxidation states (+2 and +3). In the +2 state, it forms a wide range of ionic compounds including the oxide, hydroxide and halides. In the +3 oxidation state, it forms a wide range of complexes. [Reference 1, p. 58]	
Source of Radioactivity (This should be back to the point of generation.)		The coolant is subjected to bombardment by neutrons as it flows through the reactor, and these neutrons can activate some trace impurities from stable atoms to being radioactive. Cobalt-58 is produced by neutron activation of Nickel-58, which is a major constituent of the steam generator tubes and the stainless steel in core and vessel materials. Steel activation products are released into the coolant by a variety of mechanisms including corrosion. [Reference 2, p. 24] In the primary coolant system, the corrosion product activities including Co-58 can be transported throughout the system in either soluble forms or insoluble crud.	
Source Activity (Bq) (before abatement)		**Design Basis** Reactor Coolant Activity: 1.9E-03 µCi/g [70.3Bq/g] [Reference 3, Table 11.1-2] **Realistic Source Terms** Reactor Coolant Activity: 3.9E-03 µCi/g [144.3Bq/g] [Reference 3, Table 11.1-8]	
Radionuclide or Group of Radionuclides	Co-60	**Discharge Route (Solid/Liquid/Gas)**	Gas / Liquid
Radioactive Half-life	5.27 years	**Parent**	Co-60m
		Daughter	Ni-60
Principal Decay Mode	Beta [gamma]	**Average Energy**	0.0965 MeV
		Total Gamma Energy	2.5 MeV

<table>
<tr><th colspan="4">9. BAT Assessment Form – Beta Particulates (cont.)</th></tr>
<tr><td colspan="2">Speciation</td><td colspan="2">Cobalt is a transition metal element that shows two common oxidation states (+2 and +3).
In the +2 state, it forms a wide range of ionic compounds including the oxide, hydroxide and halides.
In the +3 oxidation state, it forms a wide range of complexes.
[Reference 1, p. 60]</td></tr>
<tr><td colspan="2">Source of Radioactivity
(This should be back to the point of generation.)</td><td colspan="2">Cobalt-60 is produced by neutron activation of the stable reactor steel component Cobalt-59 in the hard-wearing alloy, Stellite. [Reference 2, p. 24]
The reactor coolant contains Cobalt-59 as a circulating corrosion product that is released from reactor system surfaces. This deposits on the fuel cladding as crud and becomes activated Cobalt-60. Cobalt-60 is then released into the coolant and deposits onto other RCS surfaces.</td></tr>
<tr><td colspan="2">Source Activity (Bq)
(before abatement)</td><td colspan="2">Design Basis
Reactor Coolant Activity: 2.2E-04 µCi/g [8.14 Bq/g]
[Reference 3, Table 11.1-2]
Realistic Source Terms
Reactor Coolant Activity: 4.4E-04 µCi/g [16.28 Bq/g]
[Reference 3, Table 11.1-8]</td></tr>
<tr><td>Radionuclide or Group of Radionuclides</td><td>Fe-55</td><td>Discharge Route (Solid/Liquid/Gas)</td><td>Gas / Liquid</td></tr>
<tr><td>Radioactive Half-life</td><td>2.68 years</td><td>Parent
Daughter</td><td>N/A
Mn-55</td></tr>
<tr><td>Principal Decay Mode</td><td>X Ray</td><td>Average Energy</td><td>0.0965 MeV</td></tr>
<tr><td colspan="2">Speciation</td><td colspan="2">-</td></tr>
<tr><td colspan="2">Source of Radioactivity
(This should be back to the point of generation.)</td><td colspan="2">The coolant is subjected to bombardment by neutrons as it flows through the reactor, and these neutrons can activate some trace impurities from stable atoms to become radioactive.
Iron-54 is formed from reactor material corrosion and/or wear products and is deposited on the fuel surfaces or the in-core structure materials. This may be activated by neutrons to form Iron-55 and is subsequently released and transported to the radwaste system.
Under normal operating conditions, among common activated corrosion products, Iron-55 is found to be truly insoluble.</td></tr>
</table>

9. BAT Assessment Form – Beta Particulates (cont.)			
Source Activity (Bq) (before abatement)		**Design Basis** Reactor Coolant Activity: 5.0E-04 μCi/g [18.5Bq/g] [Reference 3, Table 11.1-2] **Realistic Source Terms** Reactor Coolant Activity: 1.0E-03 μCi/g [37 Bq/g]] [Reference 3, Table 11.1-8]	
Radionuclide or Group of Radionuclides	**Ni-63**	**Discharge Route (Solid/Liquid/Gas)**	Gas / Liquid
Radioactive Half-life	100 years	**Parent** **Daughter**	N/A Cu-63
Principal Decay Mode	Beta	**Average Beta Energy**	0.017 MeV
Speciation		Nickel is a transition metal element that shows two common oxidation states. The +2 state is the most stable in terms of the properties of the compounds for variations in pH and Eh. Such compounds include the halides, hydroxide and carbonate.	
Source of Radioactivity (This should be back to the point of generation.)		The coolant is subjected to bombardment by neutrons as it flows through the reactor, and these neutrons can activate some trace impurities from stable atoms to being radioactive. Nickel-62 is formed from reactor material corrosion and/or wear products and is deposited on the fuel surfaces or the in-core structure materials. This may be activated by neutrons to form Nickel-63 and is subsequently released and transported to the radwaste system.	
Source Activity (Bq) (before abatement)		N/A	

9. BAT Assessment Form – Beta Particulates (cont.)	
Techniques to Prevent or Minimise at Source (options considered and selected)	**Materials Selection.** Control of the choice of materials in contact with the primary coolant leads to a reduction in the production of corrosion products including Co-58, Co-60, Fe-55 and Ni-63. In the AP1000 NPP design, the following points are of particular note: • The parts of the control rod drive mechanisms and control rod drive line exposed to reactor coolant are made of metals that resist the corrosive action of the coolant, thereby reducing the amount of radioactivity carried by the coolant. Three types of metals are used exclusively: stainless steels, nickel-chromium-iron alloys, and, to a limited extent, cobalt-based alloys. In the case of stainless steels, only austenitic and martensitic stainless steels are used. [Reference 3, subsection 4.5.1.1] • Ferritic low-alloy and carbon steels used in principal pressure-retaining applications have corrosion-resistant cladding on surfaces exposed to the reactor coolant. The corrosion resistance of the cladding material is at least equivalent to the corrosion resistance of Types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron alloy, martensitic stainless steel, and precipitation-hardened stainless steel. [Reference 3, subsection 5.2.3.2.2] • Hardfacing material in contact with reactor coolant is primarily a qualified low or zero cobalt alloy equivalent to Stellite-6. The use of cobalt base alloy is minimised. Low or zero cobalt alloys used for hardfacing or other applications where cobalt alloys have been previously used are qualified using wear and corrosion tests. The corrosion tests qualify the corrosion resistance of the alloy in reactor coolant. Cobalt free wear resistant alloys considered for this application include those developed and qualified in nuclear industry programs. [Reference 3, subsection 5.2.3.2.2] • Equipment specifications for components exposed to high temperature reactor coolant contain limitations on the cobalt content of the base metal as given in Table 12.3-1 of the DCD. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations. Nickel-based alloys in the RCS (Co-58 is produced from activation of Ni-58) are similarly used only where component reliability may be compromised by the use of other materials. [Reference 3, subsection 12.3.1.1.1] • The specification of low cobalt tubing material for the AP1000 NPP steam generator design is an important feature of the design; not only in terms of reduced exposure relative to the steam generator, but to the total plant radiation source term. The cobalt content has been substantially reduced to 0.015 weight percent for the AP1000 NPP steam generator tubing. [Reference 3, subsection 12.3.1.1.1]

9. BAT Assessment Form – Beta Particulates (cont.)	
Techniques to Prevent or Minimise at Source (cont.) (options considered and selected)	**Materials Quality Control** To ensure low corrosion rates in the AP1000 NPP, good quality assurance and quality control systems need to be implemented during manufacture and construction. According to the DCD, austenitic stainless steel materials used in the fabrication, installation, and testing of nuclear steam supply components and systems are handled, protected, stored, and cleaned according to recognized, accepted methods designed to minimise contamination that could lead to stress corrosion cracking. The procedures covering these controls are stipulated in process specifications. Tools used in abrasive work operations on austenitic stainless steel, such as grinding or wire brushing, do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or stress-corrosion cracking. [Reference 3, subsection 5.2.3.4.1] **Piping Design** The piping in pipe chases is designed for 60 year design objective with consideration for corrosion and operating environment. Pipe bends are used instead of elbows where practicable to reduce potential crud traps. Welds are made smooth to prevent crud traps from forming. [Reference 3, subsection 12.3.1.1.1] **Reactor Coolant System Chemical Control** Chemical control of the reactor coolant is critical in ensuring reduction of corrosion and crud thereby reducing the amount of radioactivity carried by the coolant. In the AP1000 NPP the following measures are employed: • The RCS water chemistry is selected to minimise corrosion. Routinely scheduled analyses of the coolant chemical composition are performed to verify that the reactor coolant chemistry meets the specifications. Other additions, such as those to reduce activity transport and deposition, may be added to the system. • The CVS provides a means for adding chemicals to the RCS. The chemicals perform the following functions: – Control the pH of the coolant during pre-startup testing and subsequent operation – Scavenge oxygen from the coolant during heatup – Control radiolysis reactions involving hydrogen, oxygen, and nitrogen during power operations following startup • Suspended solid (corrosion product particulates) and other impurity concentrations are maintained below specified limits by controlling the chemical quality of makeup water and chemical additives and by purification of the reactor coolant through the CVS [Reference 3, subsection 5.2.3.2.1]

9. BAT Assessment Form – Beta Particulates (cont.)	
Techniques to Prevent or Minimise at Source (cont.) (options considered and selected)	The chemical treatment of primary coolant is optimised to reduce corrosion rates. The RCS water chemistry is controlled to minimise corrosion by the addition of chemicals using the CVS. Several methods are employed as follows: • A constant elevated pH value is maintained in the primary coolant by optimised regulation of the lithium concentration. [Reference 3, subsection 9.3.6.2.3.2] This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/nickel-chromium-iron systems. [Reference 3, subsection 5.2.3.2.1] • During plant startup from cold shutdown, an oxygen scavenging agent (hydrazine) is introduced. [Reference 3, subsections 9.3.6.2.4.1, 5.2.3.2.1] • During power operations, dissolved hydrogen is added to the RCS to eliminate free oxygen produced by radiolysis in the core and to prevent ammonia formation. This reduces the oxygen content and limits radiolysis. [Reference 3, subsection 9.3.6.2.4.2, 5.2.3.2.1] • Relatively little boric acid is used during power operation, since load follow is accomplished with gray rods and without changes in the RCS boron concentration. Therefore, the boric acid which is injected has a negligible effect on the free oxygen level in the RCS. [Reference 3, subsection 9.3.6.3.3] • Zinc injection into the primary system [Reference 3, subsection 9.3.6.2.3.2 and Reference 8, subsection 9.5.9.4]. This means: - Corrosion films become thinner but more stable, reducing ongoing corrosion of reactor vessel materials. - Divalent cations are displaced, released into the coolant, and blocked from redeposition. - The risk of a crud induced power shift (CIPS) is reduced.[Reference 9, Slide 41]

9. BAT Assessment Form – Beta Particulates (cont.)

Techniques for minimisation at source – Optioneering

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT.)

Criteria (Scoring -2 to 2, with 2 good and-2 poor, 0 indicates neither a benefit nor a disadvantage)	**Technique for minimising production of particulate** (Green indicates techniques employed in AP1000 NPP)					
	Material Selection and QA/QC	**Constant Elevated pH**	**Hydrazine Addition**	**Oxygen Elimination**	**Zinc Injection**	**Piping design**
Proven Technology	2	2	2	2	2	2
Available Technology	2	2	2	2	2	2
Effective Technology	2	2	2	2	2	2
Ease of Use	2	1	1	1	1	2
Cost	-2	-1	-1	-1	-1	-1
Impact (Public Dose)	2	2	2	2	2	2
Impact (Operator Dose)	2	2	2	2	2	2
Impact (Environmental)	2	2	2	2	2	2
Generates Suitable Waste Form	2	2	2	2	2	2
Secondary & Decommissioning Waste	2	2	2	2	2	2
Totals	16	16	16	16	16	17

Notes on scoring of minimisation techniques

No comments

Pathway to Environment from Source

Activated corrosion products are released from the fuel surface deposits by erosion and spalling caused by hydraulic shear forces in some cases and by dissolution in other cases. Some activated products are released from in-core materials by dissolution and wear. The activation products in the coolant can be soluble or insoluble, and they are transported by water to all parts of the primary system. This presents problems with regard to accessibility and safe maintenance of various components because of radiation fields. Among the activated corrosion products, γ-emitting activities such as Co-60 are more significant in creating the radiation field problems. Fe-55 and Ni-63 are longer-lived species and thus creates problems with radioactive waste handling and disposal. [Reference 7, p. 69]

The corrosion product transport in the PWR primary system is a continuous process of crud transport from one surface to another via the primary coolant. The crud can be quite mobile, and the major factors affecting crud transport (deposition/ dissolution) are believed to be the coolant pH and the hydrogen concentration. [Reference 7, Section 4.3.1, p. 98]

<table>
<tr><th colspan="2">9. BAT Assessment Form – Beta Particulates (cont.)</th></tr>
<tr><td>Pathway to Environment from Source (cont.)</td><td>
<p>Cobalt-58 and 60 build up in the reactor cooling water and create a problem if there are leaks and when components are opened up for maintenance. They enter the WLS during such maintenance or as a result of plant operations such as spent filter or ion exchange resin processing and decontamination of reactor components.</p>
<p>Cobalt-58 and 60 are found in either soluble cationic or insoluble forms, depending on the iron crud concentration in reactor water. They are a major constituent of liquid radwaste and even after processing can be measured in liquid effluent. They are relatively insoluble and their half-lives are relatively short, but contribution to doses to members of the public is significant. [Reference 2, subsection 2.2.5, 2.2.6]</p>
<p>Iron-55 can be found in working areas of the power station and therefore is detectable in solid waste. However it has very little potential for creating radiation dose and so it is not significant in gaseous or liquid discharges. [Reference 2, subsection 2.2.8]</p>
<p>The main mechanisms for the potential release of beta particulates to the environment are as follows:</p>

Liquid drops carried in the gaseous waste stream as an aerosol
Re-suspension of radioactivity as dust from surfaces where coolant has dried-out

<p>Any particles that are discharged into the air outside the station may be deposited either directly onto vegetation or onto the ground from where plants may absorb and to an extent concentrate the radioactivity. The principal exposure pathway is through external irradiation. Under normal circumstances, inhalation and ingestion pathways are not significant. The measured discharges of beta emitting particulate material confirms that the potential doses from them to even the potentially most exposed members of the public are low. [Reference 2, subsection 2.2.1]</p>
<p>Liquid Pathway</p>
<p>Around 1000 m^3 of reactor coolant is discharged (after processing) each year [Reference 3, Table 11.2-1], accounting for the majority of particulate discharges, the remainder being discharged to atmosphere or becoming incorporated in solid waste. This can be seen by comparing gaseous and liquid discharges of particulate.</p>
<p>Gaseous Pathway</p>
<p>The gaseous discharge route is described in Figures 11.3-1 and 11.3-2 of the AP1000 NPP DCD [Reference 3]. Airborne particulate is discharged to the atmosphere via the main ventilation stack.</p>
</td></tr>
</table>

9. BAT Assessment Form – Beta Particulates (cont.)	
Downstream Abatement Techniques (options considered and selected)	Downstream abatement techniques include consideration of the following: **a) Maintenance of dissolved oxygen and H_2O_2 concentrations in the coolant to ensure that corrosion products remain dissolved since:** • This prevents particulate crud releases. • Dissolved activated corrosion products are easily cleaned up by the CVS [Reference 9, Slide 27]. Note – The RCS water chemistry is selected to minimise corrosion. Routinely scheduled analyses of the coolant chemical composition are performed to verify that the reactor coolant chemistry meets the specifications. Other additions, such as those to reduce activity transport and deposition, may be added to the system. The CVS provides a means for adding chemicals to the RCS. The chemicals perform the following functions: • Control the pH of the coolant during pre-startup testing and subsequent operation • Scavenge oxygen from the coolant during heat-up • Control radiolysis reactions involving hydrogen, oxygen, and nitrogen during power operations following startup [Reference 3, subsection 5.2.3.2.1] **b) Removal of particulates and dissolved material** can be achieved by maximising the clean-up of corrosion products during operation, shutdown, refuelling and restart, including: • Maximisation of the CVS flowrate • Mixed bed demineralisers are provided in the purification loop to remove ionic corrosion products • Monitoring of effluent and change out of filters and resin as required • Implementation of ultrasonic fuel cleaning [Reference 9, Slide 38] **c) Prevention of corrosion products depositing on the fuel**, and instead on low dose rate, out-of-core surfaces. [Reference 2, p. 27, p. 140, item 3] **d) Steam generator pre-passivation** to: • Develop a single, chromium-rich layer, eliminating the active nickel/iron fraction • Reduce corrosion product release and activation • Reduce ex-core deposition of activated corrosion products • Mitigate crud-induced fuel problems (CIPS/AOA) • Enhance effectiveness of zinc addition [Reference 9, Slide 69]

9. BAT Assessment Form – Beta Particulates (cont.)	
Downstream Abatement Techniques (cont.) (options considered and selected)	**e) Gaseous Filtration** on discharge outlets to atmosphere minimises emissions. [Reference 2, subsection 2.2.1] **f) Liquid Filtration.** After deionisation the coolant water passes through an after-filter where radioactive particulates and resin fines are removed. [Reference 3, subsection 11.2.2.1.1] **g) The Fuel Storage Pond Cooling and Clean-up System (FSPCCS)** is designed to control contamination of the Fuel Storage Pond and ensure that heat from the fuel is removed. The water is re-circulated to maintain the required chemical and radioactivity conditions, so the radioactivity transferred to the LRWS for discharge to the environment is minimised. For the radioactivity that originates from the ponds, minimisation of waste relates initially to the minimisation of contamination of the pond water, and its discharge. The fuel storage pond water chemistry is controlled to minimise fuel-clad corrosion. All these measures minimise the release of radioactivity into the pond water. [Reference 2, subsection 2.5.4] **h) Reducing unplanned outages** – Should the reactor shutdown or reduce power, boric acid has to be added to reduce the rate of the nuclear reaction. This causes a substantial letdown diversion as does returning to power, which requires the boric acid to be diluted until the reactor achieves a sustainable nuclear reaction. The increased letdown diversion causes a short-term increase in the volume of liquid effluent that requires processing. Furthermore, by diluting the corrosion products in the reactor coolant the chemical equilibrium between corrosion products in solution and those deposited on RCS surfaces is upset, causing more of the deposited corrosion products to enter solution. This can cause a short-term increase in the amount of radioactive activation products discharged to the environment. Hence, there are environmental and commercial reasons for wanting to avoid unnecessary reactor shutdowns. [Reference 2, subsection 8.2.4] **i) Monitoring of Discharges.** All liquid waste releases are monitored by a radiation monitor prior to discharge. The monitor is located on the common discharge line downstream of the WLS monitor tanks limits for radionuclide concentrations in liquid effluents discharged into unrestricted areas. These radiation monitors will provide a signal to terminate liquid radwaste releases if the discharge concentration in the line exceeds a predetermined set point. [Reference 9]

9. BAT Assessment Form – Beta Particulates (cont.)

Abatement Technique Optioneering (Particulate in Liquid)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT)

Criteria (Scoring -2 to 2, with 2 good and -2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For Particulate in Liquid** (Green indicates techniques employed in AP1000 NPP)									
	Flocculation	**Particulate separation**	**Effluent monitoring**	**Direct Discharge**	**Evaporators**	**Precipitation/ Filtration**	**Hydrocyclone**	**Mixed bed demineralsier**	**Ultrasonic fuel cleaning**	**Minimise plant shutdowns**
Proven Technology	2	-2	2	2	-1	-2	2	2	2	2
Available Technology	1	-2	2	2	2	-2	2	2	2	2
Effective Technology	-1	-2	2	-2	-2	-2	-2	2	2	2
Ease of Use	-2	-2	1	2	-2	-2	-2	-1	-1	1
Cost	-1	-1	-1	2	-2	0	-2	-1	-1	1
Impact (Public Dose)	1	0	1	-2	1	0	0	1	1	1
Impact (Operator Dose)	-1	0	1	0	-2	0	-1	-1	1	1
Impact (Environmental)	1	0	1	-2	1	0	0	1	1	1
Generates Suitable Waste Form	1	0	1	-2	0	0	0	1	1	1
Secondary & Decommissioning Waste	-2	-1	0	0	-2	-1	-1	-1	2	1
Totals	-1	-10	10	0	-7	-9	-4	5	10	13

Notes on (Liquid) Abatement Techniques

a) Evaporation – experience with evaporators has been problematic – considered that drawbacks outweigh the benefits.

b) Plant operation can significantly affect the amounts of corrosion products – therefore good plant design, optimising plant availability, good training of operators etc. are relevant contributors to minimisation of corrosion products.

9. BAT Assessment Form – Beta Particulates (cont.)

Abatement Technique Optioneering (Particulate in gas)

(Scoring the screened options against the listed criteria to justify that the chosen option(s) is BAT)

Criteria (Scoring -2 to 2, with 2 good and-2 poor, 0 indicates neither a benefit nor a disadvantage)	**Abatement Technique For Particulate in Gas** (Green indicates techniques employed in AP1000 NPP)						
	Wet Scrubbing	**Direct Discharge**	**Carbon Delay Beds**	**HEPA Filtration**			
Proven Technology	-1	2	2	2			
Available Technology	0	2	2	2			
Effective Technology	-2	-2	2	2			
Ease of Use	-2	2	2	1			
Cost	-2	2	-2	-1			
Impact (Public Dose)	0	-2	1	1			
Impact (Operator Dose)	0	1	0	-1			
Impact (Environmental)	0	-2	1	1			
Generates Suitable Waste Form	0	0	-1	-1			
Secondary & Decommissioning Waste	-1	0	-1	-1			
Totals	-8	3	6	5			

Notes on (Gaseous) Abatement Techniques

a) Carbon Delay Beds – Carbon delay beds provide an effective deep bed filter for particulate removal

b) HEPA filtration is not considered necessary after the carbon delay beds provide adequate filtration

c) HEPA filters are used to remove radioactive particulate from the radiologically controlled area ventilation system upon detection of radioactive contamination.

d) Plant operation can significantly affect the amount of particulate produced – therefore good plant design, optimising plant availability, good training of operators etc. are relevant contributors to minimisation of particulate.

RP Predicted Normal Emission

(including allowance for normal operational fluctuation)

The expected particulate releases from an AP1000 NPP are:

To atmosphere

Co-58	8.5E-03 GBq/y
Co-60	3.2E-03 GBq/y
Fe-55	n/a
Ni-63	n/a

[Reference 10, 5.2.2, Table 5-18]

9. BAT Assessment Form – Beta Particulates (cont.)	
RP Predicted Normal Emission (cont.) (including allowance for normal operational fluctuation)	**As liquid effluent** Co-58 4.1E-01 GBq/y Co-60 2.3E-01 GBq/y Fe-55 4.9E-01 GBq/y Ni-63 5.4E-01 GBq/y [Reference 10, 5.2.1, Table 5-12 and Reference 5, Table 3.4-6]
RP Predicted Maximum Emission (TBq/y) (including allowance for normal operational fluctuation)	Worst case plant discharge to the atmosphere of 'Beta Particulates' is 2.84E-05 TBq/y [Reference 5, Table 6.1-5] Worst case plant discharge as liquid of 'Beta Particulates' is 5.4E-03 TBq/y [Reference 5, Table 6.1-6]. Assumption 'Beta Particulates" = (Non-tritium Isotopes – C-14) .
Comparison with Emissions from Other Nuclear Power Stations	**South Texas 1** Gaseous Co-58 3.256E-10 TBq/y Co-60 2.313E-08 TBq/y Fe-55 - GBq/y Ni-63 - GBq/y Liquid Co-58 2.287E-05 TBq/y Co-60 0.177 GBq/y Fe-55 0.168 GBq/y Ni-63 - GBq/y [Reference 4, 5.1.3, p.33, Table 5-13] **Braidwood 1** Gaseous Co-58 - GBq/y Co-60 - GBq/y Fe-55 - GBq/y Ni-63 - GBq/y [Reference 4, 5.1.3, p. 34, Table 5-14]

9. BAT Assessment Form – Beta Particulates (cont.)	
Comparison with Emissions from Other Nuclear Power Stations (cont.)	Liquid Co-58 0.622 GBq/y Co-60 1.536 GBq/y Fe-55 0.585 GBq/y Ni-63 - GBq/y **Cook 1** Gaseous Co-58 - GBq/y Co-60 540.2 Bq/y Fe-55 - GBq/y Ni-63 - GBq/y Liquid Co-58 0.342 GBq/y Co-60 0.317 GBq/y Fe-55 0.061 TBq/y Ni-63 - GBq/y [Reference 4, 5.1.3, p.35, Table 5-15] **Vogtle 1** Gaseous Co-58 0.127 MBq/y Co-60 0.203 MBq/y Fe-55 - GBq/y Ni-63 - GBq/y Liquid Co-58 0.492 GBq/y Co-60 0.773 GBq/y Fe-55 1.380 GBq/y Ni-63 - GBq/y [Reference 4, 5.1.3, p. 36, Table 5-16]

<table>
<tr><th colspan="2">9. BAT Assessment Form – Beta Particulates (cont.)</th></tr>
<tr><td>Comparison with Emissions from Other Nuclear Power Stations (cont.)</td><td>Sizewell B - Predicted
Gaseous
Co-58 2.00E+07GBq/y
Co-60 - GBq/y
Fe-55 - GBq/y
Ni-63 - GBq/y
Liquid
Co-58 - GBq/y
Co-60 - GBq/y
Fe-55 - GBq/y
Ni-63 - GBq/y
Note – Predicted liquid discharges from Sizewell ’B’ described as ’Other radionuclides excluding tritium’ – 1.00E+08 Bq/y
[Reference 4, 5.1.3, p. 37, Table 5-17, and Reference 2]

Average normalised release 1990 – 1994 and 1995 – 1997
<table>
<tr><th></th><th>Unit</th><th>All PWR</th><th>All GCR</th><th>All BWR</th></tr>
<tr><td>1990 - 1994</td><td>GBq/ GWa</td><td>0.18</td><td>0.3</td><td>178</td></tr>
<tr><td>1995 - 1997</td><td>GBq/GWa</td><td>0.13</td><td>0.17</td><td>351</td></tr>
</table>
[Reference 6 – Table 34]</td></tr>
<tr><td>Associated Beta Particulate Dose (µSv/y)</td><td>Marine discharge – fisherman family dose 0.68E+00 µSv/y
[Reference 5, Table 5.2.12]
Aerial discharge – local resident family dose 4.89E-04 µSv/y
[Reference 5, Table 5.2.16]</td></tr>
<tr><td>Beta Particulate Dose as a Percentage of Total Dose</td><td>Marine discharge – fisherman family dose ~30%
Aerial discharge – local resident family dose ~0.02%</td></tr>
</table>

9. BAT Assessment Form – Beta Particulates (cont.)

References

1. “Radionuclides Handbook,” R&D Technical Report P3-101/SP1b, Environment Agency, October 2003.
2. “Review of the Control and Impact of the Discharge and Disposal of Radioactive Waste at Sizewell B Power Station,” SZB/THR/042, Revision 000, British Energy Generation Ltd.
3. EPS-GW-GL-700, Rev 1, “AP1000 European Design Control Document,” Westinghouse Electric Company LLC, 2009.
4. APP-WLS-M3C-049, Revision 2 “Monthly Radiation Emissions from Radioactive Nuclides,” AP1000 Calculation Note (proprietary / protect commercial) Westinghouse Electric Company LLC, 2009.
5. UKP-GW-GL-790, Rev 4, “UK AP1000 Environment Report,” Westinghouse Electric Company LLC, 2009.
6. “UNSCEAR 2000 Report,” Vol 1, Annex C. United Nations Scientific Committee on the Effects of Atomic Radiation.
7. “Radiochemistry in Nuclear Power Reactors.” Commission on Physical Sciences, Mathematics and Applications, 1996.
8. UKP-GW-GL-793, Rev 0, “AP1000 Pre-Construction Safety Report,” Westinghouse Electric Company LLC, 2011.
9. “AP1000 Water Chemistry,” K. Newmyer, Westinghouse Electric Company Presentation, 23 July 2008.
10. APP-WLS-M3C-040, Revision 0, “Expected Radioactive Effluents Associated with Advanced Plant Designs,” AP1000 Calculation Note, Westinghouse Electric Company LLC, 2008.

APPENDIX B
UTILITY PRESENTATIONS ON RADWASTE TREATMENT PRACTICES IN EUROPE

1. Operational Experience of Waste Handling at Ringhals NPP in Sweden, Vattenfall, Sweden
2. Radioactive Waste Categories According to the End-point, Iberdrola, Spain
3. Nuclear Waste in Germany. Waste Arisings, Conditioning, Storage and Final Disposal. RWE Germany